LIST OF EFFECTIVE PAGES

CHAPTER 3

DESIGN OF STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

Page	Amendment
3-1	28
3-2	28
3-3	28
3-4	28
3-5	28
3-6	28
3-7	28
3-8	28
3-9	28
3-10	28
3-11	28
3-12	28
3-i	28
3-ii	3
3-iii	28
3-iv	27
3-v	28
3-vi	26
3-vii	22
3-viii	28
3-ix	28
3-x	22
3-xi	28
3-xii	27
3-xiii	2
3-xiv	22
3-xv	22
3-xvi	22
3-xvii	3
3-xviii	26
3-xix	23
3-xx	0
3-xxi	0
3-xxii	21
3-xxiii	22
3-xxiv	22
3-xxv	22
3-xxvi	22
3-xxvii	22
3-xxviii	3
3-xxix	22
3.1-1	22
3.1-2	3
3.1-3	28
3.1-4	25
3.1-5	27
3.1-6	16
3.1-7	0
3.1-8	21
3.1-9	26
3.1-10	25
3.1-11	15
3.1-12	28
3.1-13	24
3.1-14	0
3.1-15	0
3.1-16	0
3.1-17	15
3.1-18	25
3.1-19	3
3.1-20	26
3.1-21	20
3.1-22	24
3.1-23	22
3.1-24	17
3.1-25	16
3.1-26	22
3.1-27	22
3.1-28	22
3.1-29	0
3.1-30	17
3.1-31	0
3.1-32	0
3.1-33	0
3.1-34	16
3.1-35	0
3.1-36	0
3.1-37	25
3.1-38	23
3.1-39	26
3.1-40	28
3.2-1	24
3.2-2	24
3.2-3	23
3.2-4	25
3.2-5	17
3.2-6	20
3.2-7	21
3.2-8	25
3.2-9	25
3.2-10	22
3.2-11	27
3.3-1	25
3.3-2	0
3.3-3	18

Page	Amendment
3.3-4	25
3.3-5	0
3.4-1	27
3.4-2	27
3.4-3	9
3.4-4	25
3.4-5	8
3.4-6	26A
F3.4-1	7
F3.4-2	15
F3.4-3	15
F3.4-4	15
F3.4-5	23
3.5-1	0
3.5-2	15
3.5-3	26
3.5-4	19
3.5-5	2
3.5-6	26
3.5-7	26
3.5-8	26
3.5-9	26
3.5-10	26
3.5-11	26
3.5-11a	26
3.5-12	26
3.5-13	25
3.5-14	18
3.5-15	26
3.5-16	26
3.5-17	0
3.5-18	15
3.5-19	0
3.5-20	0
3.5-21	26
3.5-22	26
F3.5-1	26
F3.5-2	26
F3.5-3	26
F3.5-4	26
F3.5-5	26
F3.5-6	15
3.6-1	28
3.6-1a	18
3.6-2	28
3.6-2a	15
3.6-3	23
3.6-4	13

Page	Amendment
3.6-5	18
3.6-6	18
3.6-7	23
3.6-8	0
3.6-9	15
3.6-10	16
3.6-11	0
3.6-12	18
3.6-13	0
3.6-14	0
3.6-15	18
3.6-16	18
3.6-17	0
3.6-18	0
3.6-19	0
3.6-20	18
3.6-21	0
3.6-22	18
3.6-23	19
3.6-24	0
3.6-25	0
3.6-26	0
3.6-27	0
3.6-28	25
3.6-29	19
3.6-30	23
3.6-31	23
3.6-32	22
3.6-32a	12
3.6-33	25
3.6-34	25
3.6-35	15
3.6-36	0
3.6-37	16
3.6-38	0
F3.6-1	0
F3.6-2	15
F3.6-3	0
F3.6-4	23
F3.6-5	0
F3.6-6	0
F3.6-7	0
F3.6-8	0
F3.6-9	0
F3.6-9A	0
F3.6-9B	0

LIST OF EFFECTIVE PAGES (Cont'd)

CHAPTER 3

Page	Amendment	Page	Amendment
F3.6-10	22	F3.6-55	0
F3.6-11	23	F3.6-56	0
F3.6-12	23	F3.6-57	0
F3.6-13	22	F3.6-58	0
F3.6-14	22	F3.6-59	0
F3.6-15	22	F3.6-60	0
F3.6-16	22	F3.6-61	0
F3.6-17	22	F3.6-62	0
F3.6-18	22	F3.6-63	0
F3.6-19	23	F3.6-64	18
F3.6-20	23	F3.6-65	0
F3.6-21	23	F3.6-66	18
F3.6-22	23	F3.6-67	0
F3.6-23	23	F3.6-68	0
F3.6-24	23		
F3.6-25	23	3.7-1	15
F3.6-26	23	3.7-2	0
F3.6-27	23	3.7-3	0
F3.6-28	23	3.7-4	26
F3.6-29	18	3.7-5	20
F3.6-30	23	3.7-6	15
F3.6-31	23	3.7-7	15
F3.6-32	23	3.7-8	26
F3.6-33	0	3.7-9	0
F3.6-34	0	3.7-10	0
F3.6-35	16	3 7-11	0
F3.6-36	16	3.7-12	0
F3.6-37	0	3.7-13	0
F3.6-38	0	3.7-14	0
F3.6-39	0	3.7-15	0
F3.6-40	0	3.7-16	15
F3.6-41	0	3.7-17	15
F3.6-42	0	3.7-17a	15
F3.6-43	0	3.7-18	15
F3.6-44	0	3.7-19	15
F3.6-45	24	3.7-20	21
F3.6-46	0	3.7-21	0
F3.6-47	0	3.7-22	21
F3.6-48	16	3.7-23	21
F3.6-49	16	3.7-24	16
F3.6-50	26	3.7-25	26
F3.6-51	24	3.7-26	3
F3.6-52	16	3.7-27	26
F3.6-53	16		
F3.6-54	0		

LIST OF EFFECTIVE PAGES (Cont'd)

CHAPTER 3

Page	Amendment	Page	Amendment
3.7-28	0	3.7-72	0
3.7-29	15	3.7-73	0
3.7-30	26	3.7-74	0
3.7-31	0	3.7-75	0
3.7-32	26	3.7-76	0
3.7-33	0	3.7-77	0
3.7-34	0	3.7-78	0
3.7-35	0	3.7-79	0
3.7-36	0	3.7-80	0
3.7-37	0	3.7-81	0
3.7-38	15	3.7-82	16
3.7-39	15	3.7-83	0
3.7-40	0	3.7-84	16
3.7-41	0	3.7-85	16
3.7-41a	26		
3.7-42	22	3.7-86	16
3.7-42a	21	3.7-87	16
3.7-43	21	3.7-88	16
3.7-44	21	3.7-89	16
3.7-45	21	3.7-90	16
3.7-46	15	3.7-91	0
3.7-47	21	3.7-92	3
3.7-48	0	3.7-93	3
3.7-49	0	3.7-94	0
3.7-50	0	3.7-95	0
3.7-51	0		
3.7-52	0	F3.7-1	0
3.7-53	0	F3.7-2	0
3.7-54	8	F3.7-3	0
3.7-55	0	F3.7-4	0
3.7-56	0	F3.7-5	0
3.7-57	0	F3.7-6	0
3.7-58	0	F3.7-7	0
3.7-59	0	F3.7-8	0
3.7-60	0	F3.7-9	0
3.7-61	0	F3.7-10	0
3.7-62	0	F3.7-11	0
3.7-63	0	F3.7-11A	0
3.7-64	0	F3.7-12	0
3.7-65	0	F3.7-13	0
3.7-66	0	F3.7-14	0
3.7-67	0	F3.7-15	0
3.7-68	0	F3.7-16	0
3.7-69	0	F3.7-17	0
3.7-70	0	F3.7-18	0
3.7-71	0	F3.7-19	0
		F3.7-20	0

LIST OF EFFECTIVE PAGES (Cont'd)

CHAPTER 3

Page	Amendment	Page	Amendment
F3.7-21	0	3.8-23	15
F3.7-22	0	3.8-24	25
F3.7-23	0	3.8-25	0
F3.7-24	0	3.8-26	0
F3.7-25	3	3.8-27	0
F3.7-26	3	3.8-28	0
F3.7-27	0	3.8-29	0
F3.7-28a	21	3.8-30	0
F3.7-28b	21	3.8-31	0
F3.7-29	3	3.8-32	0
F3.7-30	3	3.8-33	8
F3.7-31	3	3.8-34	8
F3.7-32	0	3.8-35	15
F3.7-33	3	3.8-36	15
F3.7-34	3	3.8-37	26
F3.7-35	3	3.8-38	0
F3.7-36	0	3.8-39	17
F3.7-37	0	3.8-40	26
F3.7-38	0	3.8-41	25
F3.7-39	0	3.8-42	25
F3.7-40	21	3.8-43	15
		3.8-44	0
3.8-1	23	3.8-45	20
3.8-2	20	3.8-46	23
3.8-3	20	3.8-47	14
3.8-4	20	3.8-48	0
3.8-5	0	3.8-49	0
3.8-6	22	3.8-50	0
3.8-7	20	3.8-51	0
3.8-8	0	3.8-52	0
3.8-9	20	3.8-53	0
3.8-10	0	3.8-54	0
3.8-11	25	3.8-55	0
3.8-12	25	3.8-56	25
3.8-13	20	3.8-57	25
3.8-14	25	3.8-58	0
3.8-15	0	3.8-59	17
3.8-16	0	3.8-60	0
3.8-17	0	3.8-61	15
3.8-18	0	3.8-62	25
3.8-19	19	3.8-63	25
3.8-20	23	3.8-64	25
3.8-21	26	3.8-65	15
3.8-22	25	3.8-66	0
		3.8-67	0
		3.8-68	0

CHAPTER 3

Page	Amendment
3.8-69	0
3.8-70	0
3.8-71	15
3.8-72	0
3.8-73	0
3.8-74	25
3.8-75	0
3.8-76	9
3.8-77	21
3.8-78	9
3.8-79	20
3.8-79a	20
3.8-80	20
3.8-80a	25
3.8-81	20
3.8-82	25
3.8-83	14
3.8-84	14
3.8-85	25
3.8-86	25
3.8-87	25
3.8-88	23
3.8-89	15
3.8-90	28
3.8-91	19
3.8-92	0
3.8-93	0
3.8-94	25
3.8-95	17
3.8-96	17
3.8-97	0
3.8-98	25
3.8-99	0
3.8-100	0
3.8-101	0
3.8-102	0
3.8-103	0
3.8-104	0
3.8-105	0
3.8-106	0
3.8-107	0
3.8-108	22
3.8-109	0
3.8-110	22
3.8-111	0
3.8-112	0
3.8-113	0
3.8-114	0
3.8-115	0
3.8-116	17
3.8-117	0
3.8-118	0
3.8-119	0
3.8-120	19
3.8-121	19
3.8-122	0
3.8-123	0
3.8-124	0
3.8-125	14
3.8-125a	14
3.8-126	17
3.8-127	17
F3.8-1	15
F3.8-1a	15
F3.8-2	15
F3.8-3	15
F3.8-3a	15
F3.8-4	0
F3.8-5	0
F3.8-6	0
F3.8-7	0
F3.8-8	0
F3.8-8a	0
F3.8-8aa	15
F3.8-8B	0
F3.8-9	22
F3.8-9a	22
F3.8-10	22
F3.8-11	22
F3.8-12	15
F3.8-13	15
F3.8-14	0
F3.8-15	0
F3.8-16	0
F3.8-17	0
F3.8-18	0
F3.8-19	0
F3.8-20	0
F3.8-21	0
F3.8-22	0
F3.8-23	15
F3.8-24	15

LIST OF EFFECTIVE PAGES (Cont'd)

CHAPTER 3

Page	Amendment	Page	Amendment
F3.9-1	0	3.11-35	0
F3.9-2	3	3.11-36	22
F3.9-3	3	3.11-37	22
F3.9-4	3	3.11-38	22
F3.9-5	3	3.11-39	22
F3.9-6	3	3.11-40	22
3.10-1	24	3A-1	18
3.10-2	25	3A-2	0
3.10-3	25	3A-3	0
3.10-4	25	3A-4	0
3.10-5	25	3A-5	0
		3A-6	0
3.11-1	26	3A-7	0
3.11-1a	26	3A-8	0
3.11-2	22	3A-9	0
3.11-3	0	3A-10	0
3.11-4	0	3A-11	0
3.11-5	22	3A-12	0
3.11-6	0	3A-13	17
3.11-7	26	3A-14	17
3.11-8	18	3A-15	0
3.11-9	24	3A-16	19
3.11-10	22	3A-17	6
3.11-10a	19	3A-18	19
3.11-11	16	3A-19	0
3.11-12	24	3A-20	0
3.11-13	18	3A-21	0
3.11-14	0	3A-22	0
3.11-15	0	3A-23	17
3.11-16	24	3A-24	0
3.11-17	17	3A-25	0
3.11-18	25	3A-26	0
3.11-19	24	3A-27	0
3.11-20	0	3A-28	0
3.11-21	0	3A-29	0
3.11-22	22	3A-30	0
3.11-23	22	3A-31	0
3.11-24	22	3A-32	0
3.11-25	0	3A-33	0
3.11-26	0	3A-34	0
3.11-27	2	3A-35	0
3.11-28	26	3A-36	0
3.11-29	22	3A-37	0
3.11-30	0	3A-38	0
3.11-31	0	3A-39	0
3.11-32	0	3A-40	0
3.11-33	0	3A-41	0
3.11-34	0	3A-42	0

LIST OF EFFECTIVE PAGES (Cont'd)

CHAPTER 3

Page	Amendment	Page	Amendment
3A-43	0	3C-5	17
3A-44	0	3C-6	0
3A-45	0	3C-7	0
3A-46	0	3C-8	0
3A-47	0	3C-9	27
3A-48	0	3C-10	0
3A-49	0	3C-11	0
3A-50	0	3C-12	0
3A-51	0		
		3D-1	18
3B-1	25	3D-2	17
3B-2	19	3D-3	17
3B-3	0	3D-4	0
3B-4	0	3D-5	0
3B-5	19	3D-6	20
3B-6	25	3D-7	17
3B-7	0	3D-8	17
3B-8	0	3D-9	0
3B-9	0	3D-10	18
3B-10	0	3D-11	0
3B-11	0	3D-12	0
3B-12	8		
3B-13	0		
3B-14	0	F3D-1	0
3B-15	0	F3D-2	12
3B-16	0		
3B-17	0	3E-i	0
3B-18	0	3E-1	0
3B-19	0	3E-2	0
3B-20	0	3E-3	0
3B-21	0	3E-4	0
3B-22	0		
3B-23	25	3F-1	18
3B-24	0	3F-2	0
3B-25	0	3F-3	0
3B-26	0	3F-4	0
3B-27	0	3F-5	0
3B-28	0	3F-6	7
3B-29	0	3F-7	0
3B-30	0	3F-8	0
3B-31	0	3F-9	24
3B-32	0	3F-10	18
3B-33	0	3F-11	18
3B-34	0	3F-12	7
3B-35	25	3F-13	18
3B-36	0	3F-14	18
		3F-15	23
		3F-16	22
3C-1	18	3F-17	25
3C-2	17	3F-18	0
3C-3	17	3F-19	0
3C-4	27	3F-20	22

LIST OF EFFECTIVE PAGES (Cont'd)

CHAPTER 3

Page	Amendment	Page	Amendment
3F-21	19	3G-1	26
3F-22	7	3G-2	0
3F-23	0	3G-3	0
3F-24	0	3G-4	0
3F-25	19	3G-5	0
3F-26	17	3G-6	0
3F-27	17	3G-7	0
3F-28	0	3G-8	0
3F-29	28	3G-9	0
3F-30	26	3G-10	0
3F-31	22	3G-11	0
3F-32	17	3G-12	0
3F-33	18	3G-13	0
3F-34	25	3G-14	0
3F-35	7	3G-15	0
3F-36	0	3G-16	0
3F-37	0	3G-17	0
3F-38	0	3G-18	0
3F-39	0	3G-19	0
3F-40	0	3G-20	0
3F-41	0	3G-21	0
3F-42	0	3G-22	0
		3G-23	0
F3F-1	0	3G-24	0
F3F-2	0	3G-25	0
F3F-3	0	3G-26	0
F3F-4	0	3G-27	0
F3F-5	0	3G-28	0
F3F-6	15	3G-29	0
F3F-7a	22	3G-30	0
F3F-7b	15	3G-31	0
F3F-7C	22	3G-32	0
F3F-7d	15	3G-33	0
F3F-7E	22	3G-34	0
F3F-7F	22	3G-35	0
F3F-8	22	3G-36	0
F3F-9	15	3G-37	0
F3F-10	15	3G-38	0
F3F-11	0	3G-39	0
F3F-12	0	3G-40	0
F3F-13	0	3G-41	0
F3F-14	0	3G-42	0
F3F-15	0	3G-43	0
F3F-16	0	3G-44	0
F3F-17	0	3G-45	0
F3F-18	0	3G-46	0
F3F-19	0	3G-47	0

LIST OF EFFECTIVE PAGES (Cont'd)

CHAPTER 3

Page	Amendment
3G-48	0
3G-49	0
3G-50	0
3G-51	0
3G-52	0
3G-53	0
3G-54	0
3G-55	0
3G-56	0
3G-57	0
3G-58	0
3G-59	0
3G-60	0
3G-61	0
3G-62	0
3G-63	0
3G-64	0
3G-65	0
3G-66	0
3G-67	0
3G-68	0
3G-69	0
3G-70	0
3H-i	22
3H-ii	0
3H-1	22
3H-2	0
3H-3	22
3H-4	25
3H-5	22
3H-6	25
3H-7	0
3H-8	0
3H-9	0
3H-10	0
3H-11	0
3H-12	0
3H-13	22
3H-14	0
3H-15	19
3H-16	0
3H-17	0
3H-18	0
3H-19	16
3H-19a	3
3H-19b	3
3H-19c	25
3H-20	3

Page	Amendment
3H-21	0
3H-22	0
3H-23	3
3H-24	3
F3H-1	3
F3H-2	0
F3H-3	0
F3H-4	0
F3H-5	0
F3H-6	0
F3H-7	0
F3H-8	0
F3H-9	0
F3H-10	0
F3H-11	0
F3H-12	0
F3H-13	0
F3H-14	0
F3H-15	0
F3H-16	0
F3H-17	0
F3H-18	0
F3H-19	0
F3H-20	0
F3H-21	0
F3H-22	0
F3H-23	0
F3H-24	0
F3H-25	0
F3H-26	0
F3H-27	0
F3H-28	0
F3H-29	0
F3H-30	3
F3H-31	3
F3H-32	3
F3H-33	3
F3H-34	3
F3H-35	3
F3H-36	3
F3H-37	3
F3H-38	3
F3H-39	3
F3H-40	3
F3H-41	3
F3H-42	3
F3H-43	3

LIST OF EFFECTIVE PAGES (Cont'd)

CHAPTER 3

Page	Amendment
3H-Ai	0
3H-A1	0
3H-A2	0
3H-A3	0
3H-A4	0
3H-A5	0
3H-A6	0
3H-A7	0
3H-A8	0
3H-A9	0
3H-A10	0
3H-A11	0
3H-A12	0
3H-A13	0
3H-A14	0
3H-A15	0
3H-A16	0
3H-A17	0
3H-A18	0
3H-A19	0
3H-A20	0
3H-A21	0
3H-A22	0
3H-A23	0
3H-A24	0
3H-A25	0
3H-A26	0
3H-A27	0
3H-A28	0
3H-A29	0
3H-A30	0
3H-A31	0
3H-A32	0
3H-A33	0
3H-A34	0
3I-i	18
3I-1	17
3I-2	0
3I-3	0
3I-4	0
3I-5	0
3I-6	18
3I-7	0
3I-8	0
3I-9	0
F3I-I	0
F3I-II	0

Page	Amendment
3I-10	0
3I-11	0
3I-12	0
3I-13	0
3I-14	28
3I-15	0
3I-16	0
3I-17	0
3I-18	0
3I-19	0
3I-20	0
3I-21	0
3I-22	17
3I-23	0
3I-24	0
3I-25	0
3I-26	0
3I-27	0
3I-28	0
3I-29	0
3I-30	0
3I-31	0
3I-32	0
3I-33	0
3J-i	12
3J-1	16
3J-2	12
3J-3	17
3J-4	17
3J-5	25
3J-6	17
3J-7	19

DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEM

Chapter 3

TABLE OF CONTENTS

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

CHAPTER 3

LIST OF TABLES

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

CHAPTER 3

LIST OF TABLES (Cont'd)

CHAPTER 3

LIST OF FIGURES

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

CHAPTER 3

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

CHAPTER 3

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

CHAPTER 3

DESIGN CRITERIA - STRUCTURES, COMPONENTS, EQUIPMENT AND SYSTEMS

3.1 CONFORMANCE WITH THE GENERAL DESIGN CRITERIA

The Construction Permit for the Hutchinson Island (St. Lucie Unit 1) Plant was issued on July 1, 1970 and preceded the publication of the (AEC) "General Design Criteria for Nuclear Power Plants" (10 CFR 50, Appendix A, February 20, 1971).

Presented are responses reflecting the design intent for this nuclear power plant in consideration of the General Design Criteria for Nuclear Power Plants.

3.1.1 CRITERION 1 - QUALITY STANDARDS AND RECORDS

Structures, systems and components important to safety shall be designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection and testing of structures, systems and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

DISCUSSION

All structures, systems and components of the facility are classified according to their relative importance to safety. Those items vital to safety such that their failure might cause or result in an uncontrolled release of an excessive amount of radioactive material are designated seismic Class 1. They and items of lesser importance to safety, are designed, fabricated, erected and tested according to the provisions of recognized codes and quality standards. Discussions of the applicable codes, standards, records and the quality assurance program used to implement and audit the construction and operation processes are presented in Sections 17.1 and 17.2. A complete set of facility structural, arrangement and system drawings will be maintained under the control of FP&L throughout the life of the plant. Quality assurance written data and comprehensive test and operating procedures are likewise assembled and maintained by FP&L. The classification of safety related structures, systems and components is discussed in Section 3.2.

3.1.2 CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

Structures, systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems and components shall reflect: (1) appropriate consideration of the most severe of natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed.

DISCUSSION

The structures, systems and components important to safety are designed to withstand the effects of natural phenomena without loss of capability to perform their safety functions. Natural phenomena factored into the design of plant structures, systems and components important to safety are determined from recorded data for the site vicinity with appropriate margin to account for uncertainties in historical data.

The most severe natural phenomena postulated to occur at the site in terms of induced stresses is the design basis earthquake (DBE). Those structures, systems, and components vital for the mitigation and control of accident conditions are designed to withstand the effects of a loss of coolant accident (LOCA) coincident with the effects of the DBE. Structures, systems and components vital to the safe shutdown of the plant are designed to withstand the effects of any one of the most severe natural phenomena, including flooding, hurricanes, tornadoes and the DBE.

Design criteria for wind and tornado, flood and earthquake are discussed in Section 3.3, 3.4 and 3.7 respectively.

3.1.3 CRITERION 3 - FIRE PROTECTION

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

DISCUSSION

NFPA 805 does not supersede the requirements of GDC 3, 10 CFR 50.48(a), or 10 CFR 50.48(f).Those regulatory requirements continue to apply to licensees that adopt NFPA 805. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements may be met is different than under 10 CFR 50.48(b). Specifically, whereas GDC 3 refers to structures, systems, and components (SSCs) important to safety, NFPA 805 identifies fire protection systems and features required to meet the Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is ensured, is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805. The Section 1.5.1 criteria include provisions for ensuring that reactivity control, inventory and pressure control, decay heat removal, vital auxiliaries, and process monitoring are achieved and maintained.

This methodology specifies a process to identify the fire protection systems and features required to achieve the nuclear safety performance criteria in Section 1.5 of NFPA 805. Once a determination has been made that a fire protection system or feature is required to achieve the performance criteria of Section 1.5, its design and qualification must meet any applicable requirements of NFPA 805, Chapter 3. Having identified the required fire protection systems and features, the licensee selects either a deterministic or performance-based approach to demonstrate that the performance criteria are satisfied. This process satisfies the GDC 3 requirement to design and locate SSCs important to safety to minimize the probability and effects of fires and explosions. (Reference 1)

3.1.4 CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analysis reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

DISCUSSION

Structures, systems and components important to safety are designed to accommodate the effects of and to be compatible with the pressure, temperature, humidity, and radiation conditions associated with normal operation, maintenance, testing, and postulated accidents including a LOCA, in the area in which they are located.

Due to the application of leak before break methodology to the RCS hot and cold leg piping, the dynamic effects associated with circumferential (guillotine) and longitudinal (slot) breaks do not have to be considered. A technical evaluation was performed to demonstrate that the probability of likelihood of such breaks occurring is sufficiently low that they need not be a design basis (see Reference 24 in Section 3.6).

Protective walls and slabs, local missile shielding, or restraining devices are provided to protect the containment and engineered safety features systems within the containment against damage from missiles generated by equipment failures. The concrete enclosing the reactor coolant system serves as radiation shielding and an effective barrier against internally generated missiles. Local missile barriers are provided for control element drive mechanisms. Penetrations and piping extending outward from the containment, up to and including isolation valves are protected from damage due to pipe whipping, and are protected from damage by external missiles, where such protection is necessary to meet the design bases.

Non-seismic Class I piping is arranged or restrained so that failure of any non-seismic Class I piping will neither cause a nuclear accident nor prevent essential seismic Class I structures or equipment from mitigating the consequences of such an accident.

Seismic Class I piping is arranged or restrained such that in the event of rupture of a Class I seismic pipe which causes a LOCA, resulting pipe movement will not result in loss of containment integrity or adequate engineered safety features systems operation.

The structures inside the containment vessel are designed to sustain dynamic loads which could result from failure of major equipment and piping, such as jet thrust, jet impingement and local pressure transients, where containment integrity is needed to cope with the conditions.

The external concrete shield building protects the steel containment vessel from damage due to external missiles such as tornado propelled missiles.

For those components which are required to operate under extreme conditions such as design seismic loads or containment post-LOCA environmental conditions, the manufacturers submit type test, operational or calculational data which substantiate this capability of the equipment.

Refer to Section 3.5, 3.6, 3.7.5 and 3.11 for details.

3.1.5 CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS OR COMPONENTS

Structures, systems and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

DISCUSSION

Safety related components interconnected between the two units include the condensate storage tanks, the diesel generator fuel oil system, and the Class 1E 4.16 kV switchgear (1AB and 2AB) Station Blackout cross-tie. These safety related interconnections are not normally used by both units and employ isolation devices between them. Locked closed isolation valves are provided for the AFW and diesel generator fuel oil inter-ties. The Station Blackout cross-tie has two breakers in series for isolation between the two units. The failure of equipment on one unit will not impair the ability of the counterpart on the other unit from performing its safety related function. The interconnections provide added redundancy and operational flexibility without compromising unit and system independence.

In accordance with NRC staff requirements, a missile protected inter-tie is provided between the Unit 1 auxiliary feedwater pump suction lines and the Unit 2 condensate storage tank (CST) to be used under administrative control. To add to the systems operational flexibility, the provision to supply the Unit 2 auxiliary feedwater pumps from the Unit 1 condensate storage tank is also provided. To prevent inadvertent draining of the Unit 2 CST to the Unit 1 CST, plant procedures for placing the inter-tie in service require that the Unit 1 CST outlet isolation valves be closed prior to placing the inter-tie line in service. This helps to assure that the water level in the Unit 2 CST is maintained at the minimum value required for safe shutdown.

In the unlikely event of a loss of offsite power, both St. Lucie Units 1 and 2 have their own 100 percent capacity redundant diesel generator sets which are available for safe shutdown.

In the unlikely event of a station blackout in one unit, i.e., total loss of AC power on-site and off-site, both units can be electrically connected, under administrative control, such that a diesel generator set from the non-blacked out unit is able to provide power to the minimum loads required to maintain both units in a hot standby condition.

The ultimate heat sink (a safety related structure) supplies emergency cooling water to both St. Lucie Units 1 and 2. The canal has sufficient cross-sectional water flow area to mitigate the consequences of a LOCA on one unit while safely shutting down the other unit.

THIS PAGE INTENTIONALLY BLANK

3.1.10 CRITERION 10 - REACTOR DESIGN

The reactor core and associated coolant, control and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

DISCUSSION

In ANSI-N 18.2, plant conditions have been categorized in accordance with their anticipated frequency of occurrence and risk to the public, and design requirements are given for each of the four categories. These categories covered by this criterion are Condition I - Normal Operation and Condition II - Faults of Moderate Frequency.

The design requirement for Condition I is that margin shall be provided between any plant parameter and the value of that parameter which would require either automatic or manual protective action; it is met by providing an adequate control system (refer to Section 7.7). The design requirement for Condition II is that such faults shall be accommodated with, at most, a shutdown of the reactor, with the plant capable of returning to operation after corrective action; it is met by providing an adequate protective system. (refer to Section 7.2 and Chapter 15)

Specified acceptable fuel design limits are stated in Section 4.4. Minimum margins to specified acceptable fuel design limits are prescribed in the Technical Specifications (Limiting Conditions for Operations) which support Chapters 4 and 15 of the Safety Analysis Report. The plant is designed such that operation within Limiting Conditions for Operation, with safety system settings not less conservative than Limiting Safety System Settings prescribed in the Technical Specifications, assures that specified acceptable fuel design limits will not be violated as a result of anticipated operational occurrences. During non-accident conditions, operation of the plant within Limiting Conditions for Operation ensures that specified acceptable fuel design limits are not approached within the minimum margins. Operator action, aided by the control systems and monitored by plant instrumentation, maintains the plant within Limiting Conditions for Operation during non-accident conditions.

3.1.11 CRITERION 11 - REACTOR INHERENT PROTECTION

The reactor core and associated coolant systems shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

DISCUSSION

In the power operating range, the combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor power in the power operating range is a decrease in reactivity; i.e., the inherent nuclear feedback characteristics is negative.

The reactivity coefficients are listed in Table 4.3-3 and are discussed in detail in Section 4.3.1.

3.1.12 CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

The reactor core and associated coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

DISCUSSION

Power level oscillations do not occur. The effect of the negative power coefficient of reactivity (see Criterion 11), together with the coolant temperature program maintained by control of regulating rods and soluble boron, provides fundamental mode stability. Power level is continuously monitored by neutron flux detectors (Section 7.2.1.1) and by reactor coolant temperature difference measuring devices.

3.1.13 CRITERION 13 - INSTRUMENTATION AND CONTROL

Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.

DISCUSSION

Instrumentation is provided, as required, to monitor and maintain significant process variables which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Controls are provided for the purpose of maintaining these variables within the limits prescribed for safe operation.

The principal variables and systems monitored include neutron level -(reactor power); reactor coolant temperature, flow, and pressure; pressurizer liquid level; steam generator level and pressure; and containment pressure and temperature. In addition, instrumentation is provided for continuous automatic monitoring of process radiation level in the reactor coolant system.

The following is provided to monitor and maintain control over the fission process during both transient and steady state periods over the lifetime of the core:

a) Ten independent channels of nuclear instrumentation, which constitute the primary monitor of the fission process. Of these channels, the four wide range channels are used to monitor the reactor from start-up through full power; four will monitor the reactor in the power range and are used to initiate a reactor shutdown in the event of overpower.

b) Two independent CEA Position Indicating Systems

c) A boron dilution alarm, which provides an alarm when a boron dilution event is in progress, is provided as a backup to the primary method of determining soluble poison concentration by sampling and analysis of reactor coolant water.

d) Control of reactor power by means of CEAs

e) Manual regulation of coolant boron concentrations

Incore instrumentation is provided to supplement information on core power distribution and to provide for calibration of out-of-core flux detectors.

Instrumentation measures temperatures, pressures, flows, and levels in the main steam system and auxiliary systems and is used to maintain these variables within prescribed limits.

The reactor protective system is designed to monitor the reactor operating conditions and to effect reliable and rapid reactor trip if any one or a combination of conditions deviate from a preselected operating range.

The containment pressure and radiation instrumentation is designed to function during normal operation and the postulated accidents.

The instrumentation and control systems are described in detail in Chapter 7.

3.1.14 CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture.

DISCUSSION

Reactor coolant system components are designed in accordance with the ASME Code Section III, and ANSI B 31.7. Quality control, inspection, and testing as required by these codes and allowable reactor pressure temperature operations ensure the integrity of the Reactor Coolant System.

The reactor coolant boundary is designed to accommodate the system pressures and temperatures attained under all expected modes of unit operation including all anticipated transients, and maintain the stresses within applicable stress limits.

Design pressures, temperatures and transients are listed in Chapter 5 and details of the transient analysis are provided in Chapter 15.

Means are provided to detect significant leakage from the reactor coolant pressure boundary with monitoring readouts and alarms in the control room as discussed in Chapters 5 and 12.

The pressure boundary has provisions for in-service inspection as described in Section 5.2.5, to ensure continuance of the structural and leaktight integrity of the boundary. For the reactor vessel, a material surveillance program conforming with ASTM-E-185 is provided as discussed in Chapter 5.

3.1.15 CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operations including anticipated operational occurrences.

DISCUSSION

The design criteria and bases for the reactor coolant system pressure boundary are described in the response to Criterion 14.

The operating conditions established for the normal steady state and transient operation and anticipated operational occurrences are discussed in Chapter 5. The control systems are designed to maintain the controlled plant variables within these operating limits, thereby ensuring that a satisfactory margin is maintained between the plant operating conditions and the design limits.

The reactor protective system (Section 7.2) functions to minimize the deviation from normal operating limits in the event of certain anticipated operational occurrences; the results of analyses given in Section 15.2 and 15.3 show that the design limits of the reactor coolant pressure boundary are not exceeded in the event of any anticipated operational occurrence.

3.1.16 CRITERION 16 - CONTAINMENT DESIGN

Reactor containment and associated systems shall be provided to establish an essentially leaktight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

DISCUSSION

The containment system is designed to protect the public from the consequences of a LOCA, based on a postulated break of reactor coolant piping up to and including a double ended break of the largest reactor coolant pipe.

The containment vessel, shield building, and the associated engineered safety features systems are designed to safely sustain all internal and external environmental conditions that may reasonably be expected to occur during the life of the plant, including both short and long term effects following a LOCA.

Leak tightness of the containment system and short and long term performance following a LOCA are analyzed in Section 6.2.

3.1.17 CRITERION 17 - ELECTRICAL POWER SYSTEMS

An on-site electrical power system and an off-site electrical power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The on-site electrical power sources, including the batteries, and the on-site electrical distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electrical power from the transmission network to the switchyard shall be supplied by two physically independent transmission lines (not necessarily on separate rights-of-way) designed and located so as to suitably minimize the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. Two physically independent circuits from the switchyard to the on-site electrical distribution system shall be provided. Each of these circuits shall be designed to be available in sufficient time following a loss of all on-site alternating current power sources and the other off-site electrical power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a loss of coolant accident to assure that core coolant, containment integrity, and other vital safety functions are maintained.

Provisions shall be included to minimize the probability of losing electrical power from any of the remaining sources as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the on-site electrical power sources.

DISCUSSION

Off-site power is transmitted to the plant switchyard by four physically independent 230 kv transmission lines. During normal plant operation, the station auxiliary power is normally supplied from the main generator through the plant auxiliary transformers. Upon loss of power from the auxiliary transformers, there will be a "fast dead bus" automatic transfer to the start-up transformers thus providing continuity of power.

In the event of a loss of the off-site power sources, two emergency on-site diesel generator sets and redundant sets of station batteries provide the necessary ac and dc power for safe shutdown or, in the event of an accident, provide the necessary power to restrict the consequences to within acceptable limits. The on-site emergency ac and dc power systems consist of redundant and independent power sources and distribution systems such that a single failure does not prevent the systems from performing their safety function.

Refer to Section 8.2.1 and 8.3.2 for further discussion of off-site power sources and on-site power sources respectively.

3.1.18 CRITERION 18 - INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS

Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections and switchboards to assess the continuity of the systems and the conditions of their components. The system shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as on-site power sources, relays, switches and buses, and (2) the operability of the systems as a whole, and, under conditions as close to design as practical, the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the off-site power system, and the on-site power system.

DISCUSSION

Electrical power systems important to safety are designed to permit appropriate periodic inspection and testing of important areas and features such as wiring, insulation, connections, and switchboards to assess the continuity of the systems and to detect deterioration, if any, of their components. Capability is provided to periodically test the operability and functional performance of the components of the systems. The diesel generator sets will be started and loaded periodically on a routine basis and relays, switches, and buses will be inspected and tested for operation and availability on an individual basis.

Transfers from normal to emergency sources of power will be made to check the operability of the systems and the full operational sequence that brings the systems into operation.

Refer to Sections 8.3.1.3, 8.3.2.3, and the Technical Specifications.

3.1.19 CRITERION 19 - CONTROL ROOM

A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in safe condition under accident conditions, including loss of coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent for the duration of the accident.

Equipment in appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

DISCUSSION

Following proven power plant design philosophy, all control stations, switches, controllers and indicators necessary to operate or shut down the unit and maintain safe control of the facility are located in the control room.

The design of the control room permits safe occupancy during abnormal conditions. Shielding is designed to maintain tolerable radiation exposure levels (maximum of 3 rem integrated whole body dose over a 90 day period) following design basis accidents (refer to Section 12.1). The control room will be isolated from the outside atmosphere during the initial period following the occurrence of an accident. The control room ventilation system is designed to recirculate control room air through HEPA and charcoal filters as discussed in Sections 9.4.1 and 12.2. Radiation detectors and alarms are provided. Emergency lighting is provided as discussed in Section 9.5.3.

Alternate local controls and local instruments are available for equipment required to bring the plant to and maintain a hot standby condition. It is also possible to attain a cold shutdown condition from locations outside of the control room through the use of suitable procedures. Refer to Section 7.4.1.

3.1.20 CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

DISCUSSION

The reactor protective system monitors reactor operating conditions and automatically initiates a reactor trip when the monitored variable or combination of variables exceeds a prescribed operating range. The reactor trip setpoints are selected to ensure that anticipated operational occurrences do not cause acceptable fuel design limits to be violated. Specific reactor trips are described in Section 7.2.

Reactor trip is accomplished by deenergizing the control element drive mechanism holding latch coils through the interruption of the CEDM power supply. The CEA's are thus released to drop into the core reducing reactor power.

The engineered safety features actuation system monitors potential accident conditions and automatically initiates engineered safety features and their supporting systems when the monitored variables reach prescribed setpoints. The parameters which automatically actuate engineered safety features are described in Section 7.3. Manual actuation is provided to the operator.

3.1.21 CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

DISCUSSION

The protection systems are designed to provide high functional reliability and in-service testability by designing to the requirements of IEEE 279 - 1971 and IEEE 338 - 1971. No single failure will result in the loss of the protection function. The protection channels are independent with respect to sensors and power supplies, piping, wire routing and mounting. This independence permits testing without loss of the protection function.

Each channel of the protection system, including the sensors up-to the final actuation device is capable of being checked during reactor operation. Measurement sensors of each channel used in protection systems are checked by observing outputs of similar channels which are presented on indicators and recorders in the control room. Trip units and logic are tested by inserting a signal into the measurement channel ahead of the readout and, upon application of a trip level input, observing that a signal is passed through the trip units and the logic to the logic output relays. The logic output relays are tested individually for initiation of trip action.

Protection system reliability and testability are discussed in Sections 7.2.2 and 7.3.2.

3.1.22 CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function.

DISCUSSION

The protection systems conform to the provisions of the Institute of Electrical and Electronic Engineers (IEEE) Criteria for Protection Systems for Nuclear Power Generating Stations, IEEE 279 - 1971. Four independent measurement channels complete with sensors, sensor power supplies, signal conditioning units and bistable trip units are provided for each protective parameter monitored by the protection systems. The measurement channels are provided with a high degree of independence by separate connections of the channel sensors to the process systems. Power to the channels is provided by independent emergency power supply buses.

The protective system is functionally tested to ensure satisfactory operation prior to installation in the plant. Environmental and seismic qualifications are also performed utilizing type tests and specific equipment tests. (Refer to Section 7.1.2)

3.1.23 CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air) or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

DISCUSSION

Protective system trip channels are designed to fail into a safe state or into a state established as acceptable in the event of loss of power supply or disconnection of the system. A loss of power to the CEDM holding coils results in gravity insertion of the full length CEAS into the core. Redundancy, channel independence, and separation incorporated in the protective system design minimize the possibility of the loss of a protection function under adverse environmental conditions. Refer to Sections 7.2 and 7.3.

3.1.24 CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired.

DISCUSSION

The protection systems are separated from the control instrumentation systems so that failure or removal from service of any control instrumentation system component or channel does not inhibit the function of the protection system.

3.1.25 CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITYCONTROL MALFUNCTIONS

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

DISCUSSION

Reactor shutdown with CEA's is accomplished completely independent of the control functions since the trip breakers interrupt power to the full length CEA drive mechanisms regardless of existing control signals. The design is such that the system can withstand accidental withdrawal of controlling groups without exceeding acceptable fuel design limits. Analysis of possible reactivity control malfunctions is given in Sections 15.2.1 and 15.2.2. The reactor protection system will prevent specified acceptable fuel design limits from being exceeded for any anticipated transients.

3.1.26 CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

Two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.

DISCUSSION

Two independent reactivity control systems of different design principles are provided. The first system, using control element assemblies (CEA's) includes a positive means (gravity) for inserting CEA's and is capable of controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences specified acceptable fuel design limits are not exceeded. The CEA's can be mechanically driven into the core. The appropriate margin for a stuck CEA is provided by assuming in the analyses that the highest worth CEA does not fall into the core.

The second system, using neutron absorbing soluble boron, is capable of Compensating for the rate of reactivity changes resulting from planned normal power changes, (including xenon burnout), such that acceptable fuel design limits are not exceeded.

Either system is capable of making the core subcritical from a hot operating condition and holding it subcritical in the hot standby condition. The soluble boron system is capable of holding.the reactor subcritical under cold conditions-Refer to Section 9.3.4 for details.

3.1.27 CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

The reactivity control systems shall be designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

DISCUSSION

The reactivity control systems provide the means for making and holding the core subcritical under postulated accident conditions, as discussed in Sections 9.3.4 and 4.3. Combined use of CEA's and soluble boron control by the chemical and volume control system provides the shutdown margin required for plant cooldown and long term xenon decay, assuming the highest worth CEA is stuck out of the core.

During an accident, the safety injection system functions to inject concentrated boric acid into the reactor coolant system for long term and short term cooling and for reactivity control. Details of the system are given in Section 6.3.

3.1.28 CRITERION 28 - REACTIVITY LIMITS

The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents shall include consideration of rod ejection (unless prevented by positive means) rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

DISCUSSION

The bases for control element assembly (CEA) design and control program for positioning in the core include ensuring that the reactivity worth of any one CEA is not greater than a preselected minimum value. The CEAs are divided into three sets, a shutdown set, a regulating set, and a part strength set, further subdivided into groups as necessary. Administrative procedures and interlocks ensure that only one group is withdrawn at a time, and that the regulating groups are withdrawn only after the shutdown groups are fully withdrawn. The regulating groups are programmed to move in sequence and within limits which prevent the rate of reactivity addition and the worth of individual CEAs from exceeding limiting values as discussed in Sections 4.3 and 7.1.1. The part strlength CEAs are utilized for manually controlling axial flux peaks.

The maximum rate of reactivity addition which may be produced by the chemical and volume control system is too low to induce any significant pressure forces which might degrade the reactor coolant pressure boundary leak tightness integrity or disturb the reactor vessel intervals.

The reactor coolant pressure boundary described in Chapter 5 and the reactor internals described in Chapter 4 are designed to appropriate codes delineated in the response to Criterion 14. The pressure boundary and internals can accommodate the static and dynamic loads associated with an inadvertent sudden release of energy, such as that resulting from a CEA ejection or a steam line break, without rupture and with limited deformation which will not impair the capability of cooling the core.

3.1.29 CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

The protection and reactivity control systems shall he designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

DISCUSSION

Plant conditions designated as Condition I and Condition II in ANS-N 18.2 have been carefully considered in the design of tile reactor protective system and the reactivity control systems. Consideration of redundancy, independence and testability in the design, coupled with careful component selection, overall system testing, and adherence to detailed quality assurance, assure an extremely high probability that safety functions are accomplished in the event of anticipated operational occurrences.

3.1.30 CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed, fabricated, erected and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

DISCUSSION

The reactor coolant pressure boundary components are designed, fabricated, erected and tested in accordance with the codes and standards specified in Criterion 14.

Containment sump instrumentation is used to detect reactor coolant system leakage (Section 5.2.4) by providing information on sump levels and frequency of sump pump operation. Flow instrumentation indicate and record make-up flow rate and volumes from the primary water system. This instrumentation allows detection of suddenly occurring leaks or those which are gradually increasing. Containment radiation monitors (Section 12.1) provide an additional means of reactor coolant system leakage detection.

3.1.31 CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

The reactor coolant pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady-state and transient stresses, and (4) size of flaws.

DISCUSSION

Carbon and low-alloy steel materials which form part of the pressure boundary meet the requirements of the ASME Code, Section III, paragraph N-330 at a temperature of + 40°F. The actual nil-ductility transition temperature (NDTT) of the materials has been determined by drop weight tests in accordance with ASTM-E-208. For the reactor vessel, Charpy tests will be also performed and the results will be used to plot a Charpy transition curve. The NDTT as determined by drop weight test will be used to correlate the Charpy transition curve and establish nonirradiated base points for the surveillance program. See Criterion 32 and Section 5.2.3.5.

The combined static and transient loadings are limited, whenever the reactor coolant system temperature is below NDTT + 60°F to sufficiently low values to make the probability of a rapidly propagating failure extremely remote.

All the reactor coolant pressure boundary components are constructed in accordance with the applicable codes and comply with the test and inspection requirements of these codes. These test inspection requirements assure that flaw sizes are limited so that the probability of failure by rapid propagation is extremely remote. Particular emphasis is placed on the quality control applied to the reactor vessel, on which tests and inspections exceeding code requirements are performed. The tests and inspection performed on the reactor vessel are summarized in Sections 5.4.5 and 5.4.6.

Excessive embrittlement of the reactor vessel material due to neutron radiation is prevented by providing an annulus of coolant water between the reactor core and the vessel. Results of the EPU vessel irradiation assessment are discussed in Section 4.3.2.9. Neutron fluence and radiation embrittlement of the reactor vessel are periodically monitored by the reactor vessel surveillance program described in Section 5.4.4.

A surveillance program will be conducted (see Criterion 36) to allow monitoring of the NDT temperature shift of the vessel material during its lifetime. Based on the determined NDT temperature, for a given exposure, operating restrictions to limit vessel stresses would be applied as necessary. The reactor coolant system pressure is limited at low temperatures by the pressure temperature limit curves and the low temperature overpressure protection (LTOP) system. The combination of the operating limits and the LTOP system assure that required fracture toughness requirements are maintained for the reactor coolant system pressure boundary and that stresses are sufficiently low to preclude brittle fracture.

During normal start-up for power operation, the reactor will not be made critical until the reactor coolant system temperature is at least 120°F greater than the predicted nil ductility transition temperature based on plant records of fast neutron dose to the vessel. The stress criteria include the maximum loads associated with the most severe transients during emergency conditions at operating temperature. This will assure that a reactivity-induced loading which would contribute to elastic or plastic deformation cannot occur below a reactor operating temperature corresponding to NDTT +120°F.

The activation of the safety injection systems will introduce highly borated water into the primary system at pressures significantly below operating pressures and will not cause adverse pressure or reactivity effects.

The thermal stresses induced by the injection of cold water into the vessel have been examined. Analysis shows that there is no gross yielding across the vessel wall using the minimum specified yield strength in the ASME Boiler and Pressure Vessel Code, Section III.

Adverse effects that could be caused by exposure of equipment or instrumentation to containment spray water is avoided by designing the equipment or instrumentation to withstand direct spray or by locating it or protecting it to avoid direct spray.

3.1.32 CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

Components which are part of the reactor coolant pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor pressure vessel.

DISCUSSION

Provisions are made for inspection, testing, and surveillance of the reactor coolant system boundary as described in Section 5.2.5.

The reactor vessel material surveillance program described in Section 5.4.4 conforms with ASTM-E-185-66. Sample pieces taken from the same shell plate material used in fabrication of the reactor vessel are installed between the core and the vessel inside wall. These samples will be removed and tested at intervals during vessel life to provide an indication of the extent of the neutron embrittlement of the vessel wall. Charpy tests will be performed on the samples to develop a Charpy transition curve. By comparison of this curve with the Charpy curve and drop weight tests on specimens taken at the beginning of the vessel life, the change of NDTT will be determined and operating instructions adjusted as required.

3.1.33 CRITERION 33 - REACTOR COOLANT MAKEUP

A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps and valves used to maintain coolant inventory during normal reactor operation.

DISCUSSION

Reactor coolant makeup during normal operation is provided by the chemical and volume control system (Section 9.3.4) which includes three positive displacement charging pumps rated at 44 gpm each. The design incorporates a high degree of functional reliability by provision of redundant pumps and alternate paths for charging. The charting pumps can be powered from either on-site or off-site power sources including the diesel generator sets. It is not the function of the chemical and volume control systems to provide protection against small breaks; this safety function is provided by the safety injection system. The chemical and volume control system does have the capability of balancing the flow loss to the containment for leaks in the reactor coolant boundary up to 0.30 in. equivalent diameter with only one charging pump operating. However, loss of this chemical and volume control system capability in no way compromises the safety of the reactor plant.

3.1.34 CRITERION 34 - RESIDUAL HEAT REMOVAL

A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

Residual heat removal capability is provided by the shutdown cooling system (Section 9.3-5) for reactor coolant temperatures less than 325°F. For temperatures greater than 325°F, this function is provided by the steam generators and the auxiliary feedwater system. Sufficient redundancy, interconnections, leak detection, and isolation capabilities exist in each of these systems to assure that the residual heat removal function can be accomplished, assuming a single failure. Within appropriate design limits, either system can remove fission product decay heat at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.

If onsite power is lost, there will be a "fast-dead" automatic transfer of power to the startup transformers. If offsite power is lost, the electrical equipment required for safe shutdown is loaded on the emergency diesel generators. Refer to Sections 7.4 and 8.3.2.

3.1.35 CRITERION 35 - EMERGENCY CORE COOLING

A system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

The emergency core cooling system is discussed in detail in Section 6.3.2. It consists of the high pressure safety injection subsystem and the low pressure safety injection subsystem and safety injection tanks.

The system is designed to meet the criterion stated above with respect to the prevention of fuel and clad damage that would interfere with the emergency core cooling function for the full spectrum of break sizes, and to the limitation of metal-water reaction. Each of the subsystems is fully redundant, and the subsystems do not share active components other than the valves controlling the suction headers of the high- and low-pressure safety injection pumps. Minimum safety injection is assured even though one of these valves fails to function. These valves are in no way associated with the function of the safety injection tanks.

The ECCS design satisfies the criteria specified in 10 CFR 50, Appendix K. The results of the analyses performed are given in Section 6.3.3.

3.1.36 CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit periodic inspection of important components, such as spray rings in the reactor containment vessel, water injection nozzles, and piping to assure the integrity and capability of the system.

DISCUSSION

The capability for periodic inspection of important components of the Emergency Core Cooling System (safety injection system) is provided to the extent practicable through the arrangement and location of the components of the system. System components external to the containment structure are accessible for physical inspection at any time. All components (valves and piping) inside the containment and the safety injection tanks can be inspected during refueling. See Section 6.3.

3.1.37 CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

The emergency core cooling system shall be designed to permit appropriate periodic pressure and functional testing to assure: (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection systems that transfer between normal and emergency power sources, and the operation of the associated cooling water system.

DISCUSSION

The emergency core cooling system (safety injection system) is provided with testing facilities to demonstrate system component operability. Testing can be conducted during normal plant operation with the test facilities arranged not to interfere with the performance of the systems or with the initiation of control circuits.

The safety injection system is designed to permit periodic testing of the delivery capability up to a location as close to the core as practical. Periodic pressure testing of the safety injection system is possible using the cross connection to the charging pumps in the chemical and volume control system.

The low pressure safety injection pumps are used as shutdown cooling pumps during normal plant cooldown. The pumps discharge into the safety injection header via the shutdown heat exchangers and the low pressure injection lines.

With the plant at operating pressure, operation of the safety injection pumps and valves may be verified by recirculation back to the refueling water tank. This will permit verification of flow path continuity in the high pressure injection lines and suction lines from the refueling water tank.

Borated water from the safety injection tanks may be bled through the recirculation test line to verify flow path continuity from each tank to its associated main safety injection header.

The operational sequence that brings the safety injection system into action, including transfer to alternate power sources, can be tested in parts as described in Sections 7.3.2.

3.1.38 CRITERION 38 - CONTAINMENT HEAT REMOVAL

A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss of coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for on-site electric power system operation (assuming off-site power is not available) and for off-site electric power system operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

The containment heat removal system described in Section 6.2.2 consists of the containment spray system and the containment cooling system. The containment spray system consists of two redundant subsystems each containing a containment spray pump, shutdown heat exchanger and spray header. The containment cooling system consists of four fan coolers. The containment spray system and the containment cooling system are each designed with the capacity to reduce containment pressure and temperature following a LOCA and maintain them at acceptably low levels.

Both the containment spray and the containment cooling systems are provided with emergency on-site power necessary for their operation, assuming a loss of off-site power. The systems together provide a minimum of 100 percent containment cooling capability assuming a single failure in either system or in the emergency on-site power supply.

3.1.39 CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, and piping to assure the integrity and capability of the system.

DISCUSSION

The containment spray system essential equipment except for risers, distribution header piping, spray nozzles and the containment sump strainers are located outside of the containment.
The containment sump strainers, the spray piping, and the spray nozzles within the containment can be inspected during refueling shutdowns. Associated equipment outside the containment can be visually inspected at any time.

The containment cooling system is entirely within the containment. It can be inspected at appropriate intervals during refueling shutdowns.

3.1.40 CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

The containment heat removal system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

DISCUSSION

System piping, valves, pumps, fans, heat exchangers, and other components of the containment heat removal system are designed to permit appropriate periodic testing to assure their structural and leaktight integrity. The components are arranged so that each component can be tested periodically for operability and required functional performance.

Three of four containment cooling units are normally in operation. The fourth unit will be rotated in service with the other three for normal containment cooling. Transfer to alternate power sources can be tested.

The operational sequence that would bring the containment heat removal system into action, including the transfer to alternate power sources, can be tested. With the plant at operating pressure, the containment spray pumps and valves may be operated by recirculation back to the refueling water tank. This will permit verification of flow path continuity in the suction lines from the refueling water tank to the first containment spray isolation valve outside the containment. The spray isolation valves can be tested independently of the spray pumps. Refer to Section 6.2.2.4.

3.1.41 CRITERION 41 - CONTAINMENT ATMOSPHERE CLEANUP

Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quantity of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.

Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for on-site electrical. power system operation (assuming off-site power is not available) and for off-site electrical power system operation (assuming on-site power is not available) its safety function can be accomplished, assuming a single failure.

DISCUSSION

The shield building ventilation system consists of two full capacity redundant fan and filter systems and is designed, consistent with the functioning of other engineered safety features systems, to reduce the concentration and quantity of fission products released to the environment following a LOCA by establishing and maintaining a subatmospheric pressure within the shield building to ensure that post-accident activity leakage from the containment vessel is routed through the charcoal filter system. Refer to Section 6.2.3.

Hydrogen control and sampling systems are provided to prevent the buildup of dangerous concentrations of hydrogen in the containment following a LOCA. The hydrogen control system consists of redundant hydrogen recombiners and a hydrogen purge system. The hydrogen recombiners, which are the primary means of control, provide control of hydrogen concentration in the containment without any release to the environment. The hydrogen sampling system can analyze the containment atmosphere either by passing a sample through the automatic hydrogen analyzer or by utilizing a grab sample. Containment hydrogen purging can be accomplished through the hydrogen purge system filters or by routing the purge through the shield building ventilation system filters. Refer to Section 6.2.5.

The shield building ventilation system and the containment hydrogen control system have suitable redundancy to assure that for on-site or for off-site electrical power system failure, their safety functions can be accomplished, assuming a single failure.

3.1.42 CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems.

DISCUSSION

All components of the shield building ventilation system and the containment hydrogen control system are accessible for physical inspection. Ducts, plenums, and casings are provided with access doors for internal inspection. Refer to Sections 6.2.3.4, 6.2.5.4 and the Technical Specifications.

3.1.43 CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

The containment atmosphere cleanup systems shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

DISCUSSION

The shield building ventilation system and hydrogen control and sampling systems are designed and constructed to permit periodic pressure and functional testing. For the purpose of periodically testing the retentive capability of the filter systems, test panels are placed in the filter housings in locations which allow the panels to be subjected to the same air flow as the filters. These will be periodically removed and tested.

High efficiency particulate (HEPA) and charcoal filters are located outside tile containment for convenience in testing and inspection. Periodic tests such as the following will be made:

a) Observations of differential pressure across each filter
b) Dioctylphthate (DOP) aerosol penetration tests of HEPA filters
c) Freon gas testing of charcoal filters
d) Test of a representative sample of charcoal element

Active components of the shield building ventilation system can be tested periodically for operability and required functional performance.

The full operational sequence that would bring the systems into action, including the transfer to alternate power sources, and the design air flow capability can be tested. Refer to Section 6.2.3.4 and the Technical Specifications.

3.1.44 CRITERION 44 - COOLING WATER

A system to transfer heat from structures, systems and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.

Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for on-site electrical power system operation (assuming off-site power is not available) and for off-site electrical power system operation (assuming on-site power is not available) the system safety function can be accomplished, assuming a single failure.

DISCUSSION

The cooling water systems which function to remove the combined-heat load from structures, systems and components important to safety under normal operating and accident conditions, are the component cooling system and the intake cooling water system.

The component cooling system is a closed loop system which removes heat from the shutdown heat exchangers, containment cooling system and other essential and nonessential components as described in Section 9.2.2. The system consists of three pumps with two heat exchangers, piping , valves and instrumentation arranged in two essential headers and one nonessential header. Two essential headers serve redundant safety related components. Only one essential header is needed to remove the heat generated under post-LOCA conditions.

The intake cooling water system is an open loop system which removes heat from the component cooling system and transfers it to the ultimate heat sink as described in Section 9.2.1. The system consists of three pumps with piping, valves and instrumentation arranged in two essential headers, one to each component cooling heat exchanger, and branches to two non-essential headers which supply water to the turbine cooling water heat exchangers, which are isolated automatically upon receiving SIAS. Only one essential header is needed to remove the heat generated under post-LOCA conditions.

The intake cooling water pumps normally take water from the Atlantic Ocean through the circulating water intake conduits and canal. In the event of interruption of water from this source, water is taken through the emergency cooling water canal from Big Mud Creek which serves as the ultimate heat sink. The ultimate heat sink is discussed in Section 9.2.7.

The piping, valves, pumps and heat exchangers in each system are designed and arranged so that the safety function can be performed assuming a single failure. The essential headers of each system will each be isolated from the nonessential header during the emergency mode of operation.

Electrical power for the operation of each system may be supplied from offsite or onsite emergency power sources, with distribution arranged such that a single failure will not prevent the system from performing its safety function.

3.1.45 CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

DISCUSSION

The component cooling system and intake cooling water system are designed to permit periodic inspection, to the extent practical of important components, such as heat exchangers, pumps, valves and accessible piping. Each system is normally pressurized permitting leakage detection by routine surveillance or monitoring instrumentation. Refer to Sections 9.2.1.4 and 9.2.2.4.

3.1.46 CRITERION 46 - TESTING OF COOLING WATER SYSTEM

The cooling water system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss of coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

DISCUSSION

Both the component cooling and intake cooling water systems are in operation during normal plant operation or shutdown. The structural and leaktight integrity of the component cooling and intake cooling water systems components are demonstrated in this way. Pumps and heat exchangers are operated as dictated by plant operational modes and tested on a schedule basis to monitor operational capability of redundant components. Data can be taken periodically during normal plant operation to confirm heat transfer capabilities. Refer to Sections 9.2.1.4 and 9.2.2.4.

The systems are designed to permit testing of system operability encompassing simulation of emergency reactor shutdown or LOCA conditions including the transfer between normal and emergency power sources.

3.1.50 CRITERION 50 – CONTAINMENT DESIGN BASIS

The reactor containment structure, including access openings, penetrations, and containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss of coolant accident. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

DISCUSSION

The containment structure, including access openings and penetrations, is designed to accommodate, without exceeding the design leak rate, the transient peak pressure and temperature associated with a LOCA up to and including a double ended rupture of the largest reactor coolant pipe.

The containment structure and engineered safety features systems have been evaluated for various combinations of energy release. The analysis accounts for system thermal and chemical energy, and for nuclear decay heat. The safety injection system is designed such that no single active failure could result in significant metal-water reaction. The cooling capacity of either the containment cooling system or the containment spray system is adequate to prevent over pressurization of the structure, and to return the containment to near atmospheric pressure. Refer to Section 6.2.1.

3.1.51 CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

The reactor containment boundary shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing and postulated accident conditions, and the uncertainties in determining (1) material properties (2) residual, steady-state, and transient stresses, and (3) size of flaws.

DISCUSSION

The material selected for the containment vessel is carbon steel (ASTM-A516 Grade 70) normalized to refine the grain which results in improved ductility. In addition, the actual mechanical and chemical properties of the material are documented and are within the limits for minimum ductility defined in ASTM-A516.

The containment vessel was built to Subsection B of the 1968 edition of Section III of the ASME Boiler and Pressure Vessel Code, and in accordance with this Code the materials including weld specimens were impact tested at a temperature at least 30 F below the lowest metal service temperature.

The design of the vessel reflects consideration of all ranges of temperature and loading conditions which apply to the vessel during operation, maintenance, testing and postulated accident conditions.

All seam welds in the vessel have been 100 percent radiographed and the acceptance standards of the radiographs ensured that flaws in welds did not exceed the maximum allowed by ASME Code.

Since this vessel has been post weld heat treated, residual stresses from welding will be minimal. Steady state and transient stresses have been calculated in accordance with accepted methods. Refer to Section 3.8.2.

3.1.52 CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

The reactor containment and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

DISCUSSION

The containment vessel has been designed so that initial integrated leak rate testing can be performed at design pressure after completion and installation of penetrations and equipment.

Provisions have been made in the containment design to permit periodic leakage rate tests to verify the continued leak-tight integrity of the containment. Refer to Sections 6.2.1.4 and 16.4.4.

3.1.53 CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

The reactor containment shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak tightness of penetrations which have resilient seals and expansion bellows.

DISCUSSION

The absence of insulation on the containment vessel permits periodic inspection of the exposed interior surfaces of the vessel. The lower portions of the containment vessel are totally encased in concrete and will not be accessible for inspection after the acceptance testing. It is contemplated that there will be no need for any special in-service surveillance program due to the rigorous design, fabrication, inspection and pressure testing the containment vessel receives prior to operation.

Provisions have been made to permit periodic testing at containment design pressure of penetrations which have resilient seals or expansion bellows to allow leak tightness to be demonstrated. Refer to Section 6.2.1.4.

3.1.54 CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

DISCUSSION

Piping penetrating the containment vessel shell is designed to withstand at least a pressure equal to the containment vessel maximum internal pressure. The isolation system design requires a double barrier on all of the above systems not serving accident consequence limiting systems so that no single active failure can result in loss of isolation or intolerable leakage. These lines are provided with isolation valves as indicated in Section 6.2.4.2.

Valves isolating penetrations serving engineered safety features systems will not automatically close with a containment isolation signal (CIS), but may be closed by remote manual operation from the control room to isolate any engineered safety feature when required.

Proper valve closing time is achieved by appropriate selection of valve, operator type and operator size. Refer to Table 6.2-16 for additional isolation valve information.

To ensure continued integrity of the containment isolation system, periodic closure and leakage tests shall be performed as stated in Section 6.2.4.4 and Technical Specification 3/4.6.1.

3.1.55 CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be 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:

1) one locked closed isolation valve inside and one locked closed isolation valve outside containment, or
2) one automatic isolation valve inside and one locked closed isolation valve outside containment, or
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, or
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 shall be 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.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be 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 in-service inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.

DISCUSSION

Except for the safety injection lines, the reactor coolant pressure boundary as defined in 10 CFR 50 is located within the containment. The safety injection lines are closed seismic Class I piping systems outside containment with isolation valves as indicated in Table 6.2-16. Provisions are made for leak testing as described in Section 6.2.1. Isolation valves are located as close to the containment as practical.

3.1.56 CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment shall be 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:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment, or
2) One automatic isolation valve inside and one locked closed isolation valve outside containment, or
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, or
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 shall be 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.

DISCUSSION

Lines which connect directly to the containment atmosphere and are not used to mitigate the effects of a LOCA are provided with two valves in series. The valving consists of either two automatic valves, or one automatic and one normally locked closed manual valve, or two normally closed manual valves.

Lines which connect directly to the containment atmosphere and are used for mitigating the effects of a LOCA are provided with a double containment barrier which consists of the closed piping system pressure boundary outside the containment and one isolation valve capable of remote manual actuation.

Automatic isolation valves, upon loss of power, are selected to failclose, fail-as-is, or fail-open, whichever position provides the greater safety. Isolation valves are located as close to the containment as practical. Refer to Section 6.2.4 for detailed information regarding containment isolation.

3.1.57 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 shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

DISCUSSION

Except for the shutdown cooling lines each line that penetrates the 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 located outside the containment as close to the containment as practical. Refer to Table 6.2-16.

The shutdown cooling lines arrangement of two locked closed valves, located inside containment meets the intent of GDC 57 since no single failure will prevent the recirculation of core cooling water or will adversely affect the integrity of the containment since the system is designed as seismic Class I inside containment and as a closed seismic Class I system outside containment.

3.1.60 CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design shall include 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 shall be 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.

DISCUSSION

The waste management system is described in Sections 11.2, 11.3 and 11.5, and is designed to provide controlled handling and disposal of liquid, gaseous, and solid wastes. The waste management system is designed to ensure that the general public and plant personnel are protected against exposure to radioactive material to meet the intent of 10 CFR Part 20 and 10 CFR Part 50, Appendix I (Proposed).

Liquid and gaseous radioactive releases from the waste management system are accomplished on a batch basis. All radioactive materials are sampled prior to release to ensure compliance with 10 CFR Part 20 and 10 CFR Part 50, Appendix I and to determine release rates. Radioactive materials which do not meet release limits will not be discharged to the environment. The waste management system is designed with sufficient holdup capacity and flexibility for reprocessing of wastes to ensure that releases are as low as practical.

The waste management system is designed to preclude the inadvertent release of radioactive material.

All storage tanks in the liquid waste and gaseous waste systems are administratively controlled to prevent the addition of waste to a tank which is being discharged to the environment. Each discharge path is provided with a radiation monitor which alerts plant personnel and initiates automatic closure of the isolation valves to prevent further releases in the event of noncompliance with 10 CFR Part 20 (see Section 11.4 for details).

The plant design for the handling of solid wastes is discussed in Section 11.5.

3.1.61 CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit inspection and testing of components important to safety, (2) with suitable shielding for radiation protection (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety or decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

DISCUSSION

Most of the components and systems in this category are in frequent use and no special testing is required. Those systems and components important to safety which are not normally operating will be tested periodically, e.g., temperature alarms in the fuel pool system (Section 9.1.3) and radiation alarms in the fuel pool area, and the fuel handling equipment (prior to each refueling and cask loading campaigns).

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 mr/hr during the storage period. The exposure time during refueling, and during other fuel handling operations will be limited so that the integrated dose to operating personnel does not exceed the limits of 10 CFR 20.

The waste management system (Chapter 11) is designed to permit controlled handling and disposal of liquid, gaseous, and solid wastes which will be generated during plant operation. The principal design criterion is to ensure that plant personnel and the general public are protected against exposure to radiation from wastes in accordance with limits defined in 10 CFR 20.

The fuel pool is located within the fuel handling building. The liquid waste processing equipment and the gaseous waste storage and disposal equipment are located within a separate area of the reactor 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 plant vent which is monitored.

Analysis (Section 15.4) has indicated that the accidental release of the maximum activity content of a gas decay tank will not result in doses in excess of the limits set forth in 10 CFR 100.

The fuel pool cooling system is designed to prevent damage to the spent fuel which could result in radioactivity release to the plant operating areas or the public environs.

The fuel pool is designed to withstand the postulated tornado driven missiles and seismic event without loss of the pool water.

3.1.62 CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.

The new and spent fuel storage and handling facilities are described in Sections 9.1.1 and 9.1.2. New fuel assemblies are stored in racks in parallel rows having center-to-center distances of 21 in. in both directions. New fuel is stored in air in the new fuel handling area. The high density spent fuel storage racks consist of 18 distinct modules of varying size in two regions. The cask pit rack is a Region 1 rack designed for storage of fresh or spent fuel assemblies having enrichments of up to 4.6 weight percent (w/o) U-235. Fuel assemblies are stored at a nominal 10.30 inch center-to-center spacing in the cask pit rack. Region 1 spent fuel pool storage racks are designed for storage of higher enriched irradiated fuel, with initial enrichments of up to 4.6 w/o U-235, such as might be temporarily discharged as part of a full core fuel offload. Region 1 is also designed to store fuel assemblies with enrichments up to 4.6 w/o U-235 that have not achieved sufficient burnup to be stored in Region 2. The center-to-center spacing in Region 1 is 10.12 inches. Region 2 storage cells were designed for fuel of various initial enrichments, including 4.6 w/o U-235 assemblies up to 4.6%. The center-to-center spacing in this region is 8.86 inches. The spacing is sufficient to maintain k_{eff} less than 1.0 for all the new and spent fuel assemblies when in unborated water.

3.1.63 CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

DISCUSSION

There are no residual or decay heat removal systems in the waste management system.

The fuel pool and waste management systems are provided with appropriate radiation indication and alarms. In addition, alarms are provided in the event of a reduction in fuel pool level.

3.1.64 CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss of coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences and from postulated accidents.

DISCUSSION

Provisions are made for monitoring the containment atmosphere, the facility effluent discharge paths, the operating areas within the plant and the facility environs for radioactivity that could be released from normal operation, from anticipated transients and from an accident.

Some liquid and gaseous effluents will contain radioactive matter. The waste management system function to remove radioactive material from these wastes by filtration, ion exchange or distillation prior to discharge, or to store the wastes until the radioactivity has decayed sufficiently to permit discharge.

Liquid wastes are sampled, and if the contained activity meets applicable limits, there may be released with continuous radiation monitoring to the circulating water system discharge.

Gaseous wastes are compressed and stored in the gas decay tanks. The gas is sampled to determine radioactivity concentration to assure release limits are not exceeded, and then monitoring during release through the plant vent.

The condenser air removal system discharge is monitored for gaseous activity.

Radioactive waste management and monitoring is discussed in Chapter 11. Area monitoring is discussed in Section 12.1.4.

REFERENCES

1. DBD-FP-1, Fire Protection Design Basis Document.

3.2 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

3.2.1 SEISMIC CLASSIFICATION

Seismic Class I structures, systems and components are those whose failure could cause uncontrolled release of significant amounts of radioactivity, those essential for safe shutdown of the reactor, and those whose immediate or long term operation is required following a LOCA to limit off-site exposures to values below the guidelines established for design basis accidents. When a system as a whole is designated seismic Class I, portions not associated with the safety function of the system are given no seismic classification.

The term "seismic Class I" as used herein corresponds to the term "Category I" as used in the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants", February 1972. No seismic classifications are made on the basis of plant operability hence no additional seismic classes or categories have been established.

Seismic Class I structures, systems and components have been designed to withstand loadings due to the design basis earthquake (DBE)*:

a) without loss of function or fluid boundary integrity if they are needed for safe plant shutdown or to mitigate the effects of a LOCA

b) without failure of fluid boundary integrity if such failure could result in significant uncontrolled release of radioactivity to the environment

c) without loss of function if such function is needed to detect or prevent potential significant uncontrolled release of radioactivity to the environment or if such function is needed to detect conditions requiring plant shutdown.

---

* The design basis earthquake (DBE) is that which could produce the maximum vibratory acceleration at the site as determined from evaluation of seismic and geologic information developed for the site and surrounding region. Refer to Section 2.5.

Table 3.2-1 lists seismic classifications for plant structures, systems and components. With the exception of the spent fuel pool cooling system, letdown system and the component cooling water supply to the reactor coolant pumps, the seismic classifications correspond to the seismic classifications given in AEC Safety Guide 29, "Seismic Design Classification," issued June 7, 1972, although the guide was not available at the time of issuance of the plant construction permit.

3.2.1.1 Seismic and Non-Seismic Interfaces

All interfaces between non-seismic and seismic Category I Systems are depicted on the P & I Diagram for all systems including those interfaces specifically questioned. The interfaces have been amplified for clarity. The method of protection at the interface is evident from inspection of the diagram. For clarification, some examples of interfaces of main streams are discussed below:

a) Main steam line - check valve V08117 (V08148) serves as the boundary. In addition, a seismic restraint has been provided on the piping on either side of the main steam isolation valves to ensure that the seismic forces will not affect the integrity of the valve and consequently the system.

b) Main feedwater line - check valve V09248 (V09280) serves as the interface. Seismic restraints have been provided on both sides of the valves to ensure that the seismic forces will not affect the seismic Category I sections.

c) Makeup water to condensate storage tank - the makeup water line is connected near the top of the condensate storage tank. The boundary is at the tank connection. Failure of the line will not affect the integrity of this condensate storage tank.

d) Condensate transfer pump suction line to condensate storage tank as with the makeup line, it is a low pressure line that is connected to the tank above the minimum level in the tank required to shutdown the plant. The interface is at the tank connection. Failure of this line will not affect the integrity of the tank or its ability to provide the quantity of water necessary to bring the plant to safe shutdown.

e) Intake cooling water line to turbine cooling water heat exchanger valve MV-21-2 (MV-21-3) acts as the interface between the seismic and non-seismic section. Only one valve is used at this interface. There are two lines, one to each component cooling water heat exchanger and turbine cooling heat exchanger. Only one system is required to perform under accident conditions.

f) Lubricating water line to circulating water pumps - Orifices SO-21-5A and 5B provide the boundary between the safety related ICW header and the non-safety related Circulating Water Pump lube water lines.

g) Component cooling water essential and nonessential headers - valves HCV-14-8A and HCV-14-9 provide the boundaries between the seismic header A supply and return respectively to the nonessential header. Valves HCV-14-8B and HCV-14-10 serve the same function in the B header. Because of the arrangement within the system, two valves provide isolation between the essential headers. Any single failure will not negate the ability of either redundant system.

h) Component cooling water to reactor coolant pump - valves HCV-14-1 and HCV-14-2 provide the interface for the purposes of containment isolation.

i) Makeup water to the component cooling water surge tank - makeup to the surge tank is provided from either demineralized water system (primary source of supply) or the fire protection system; these systems are interconnected upstream of the tank, located below the tank platform. The seismic interface is at the tank connection, which is at the top of the tank. Failure of either subsystem or connecting lines to the tank will not affect the integrity of the surge tank nor its ability to perform under accident conditions.

j) Primary makeup water to components inside containment (pressurizer quench tank, reactor drain tank, and reactor vessel head decontamination area) - the section of pipe which penetrates the containment is seismic Category I. Otherwise it is a non-seismic system. The seismic boundaries inside and outside containment, for the purpose of containment isolation, are check valves V15328 and V15326, respectively.

k) Primary water makeup to RWT - the makeup line to the RWT, as well as the ECCS pumps recirculation line, are connected at the top of the refueling water tank. The tank connections provide the interface. Failure of either of these lines will not affect the integrity of the RWT.

l) Steam generator blowdown lines - valves FCV-23-3, -5, -7, and -9 serve as the interface for the purpose of containment isolation for both the blowdown and sample lines.

m) Diesel oil system fill and drain lines - valves V17200 and V17211, which are normally locked closed, provide the interface for the drain lines. Four valves, V17202, V17208, V17209, and V17210, which are also normally locked closed, provide the interfaces for the fill lines.

n) Shutdown cooling to RWT recirculation lines - valves V3460 and V3459 serve as the interface, and are normally locked closed.

o) Refueling cavity supply and return lines - valves V07206 (V07189) and V07188 (V07170) normally locked closed serves as the interface in the containment penetration area,

p) Reactor cavity sump pump discharge line - valve LCV-07-11A serves as the interface for the purpose of containment isolation. Two valves in series are provided to negate the effect of single failure.

3.2.2 SYSTEM QUALITY GROUP CLASSIFICATION

Water and steam containing components (other than turbines and condensers) are designated Quality Group A,B,C or D in accordance with their importance to safety. This importance as emphasized by quality group assignment is considered in design, material, fabrication, assembly, construction and operation of the component. A single system may have components in more than one quality group.

The quality group designations are given in Table 3.2-1 for applicable components. Corresponding minimum code requirements applied to the various components in each quality group are given in Table 3.2-2. Interfaces between components of different quality groups are designated on the various system P&I diagrams at the end of chapters 5,6,9,10 and 11.

Components were assigned to quality groups on the basis of the following application criteria. These criteria correspond in general to the application criteria given on AEC Safety Guide 26, "Quality Group Classifications and Standards," issued March 23, 1972 and the safety class application criteria given in ANSI N 18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," issued January 1972. Neither of these documents was available at the time of issuance of the plant construction permit; therefore, they were not included in the specific plant design criteria.

3.2.2.1 Quality Group A

Quality Group A applies to reactor coolant pressure boundary components whose failure during normal reactor operations would prevent orderly reactor shutdown and cooldown assuming makeup is provided by normal makeup systems only.

3.2.2.2 Quality Group B

Quality Group B applies to containment vessel and to those components:

a) of the reactor coolant system not in quality Group A

b) that are necessary to:

1) remove directly residual heat from the reactor,

2) circulate reactor coolant for any safety system* purpose,

3) control within the reactor containment radioactivity released or control hydrogen in the reactor containment.

3.2.2.3 Quality Group C

Quality Group C applies to those components not in Quality Group A or B whose failure would result in significant radioactive release to the environment or that are necessary to:

a) provide or support any safety system* function

b) control accident airborne radioactivity outside the reactor containment.

3.2.2.4 Quality Group D

Quality Group D applies to those components not related to nuclear safety.

---

* A safety system (in this context) is any system that functions to shut down the reactor, cool the core or cool another safety system or the containment, and contains, controls, or reduces radioactivity released in an accident. Only those portions of the secondary systems are included (a) that are designed primarily to accomplish one of the above safety functions or (b) whose failure could prevent accomplishing one of the above functions.

TABLE 3.2-1

DESIGN CLASSIFICATIONS OF STRUCTURES, SYSTEMS AND COMPONENTS

STRUCTURE	SEISMIC CLASS	TORNADO WIND CRITERION	FLOOD CRITERION	MISSILE CRITERION	QUALITY GROUP	NOTES
Shield building	I	a	a	a	-	
Containment vessel	I	b	c	b	B	
Reactor building interior structures	I	b	c	a,b,c	-	
Reactor auxiliary building	I	a	a	a	-	
Diesel generator building	I	a	a	a (8)	-	
Intake structure	I	a	a	a	-	
Fuel handling building	I	a	a	a	-	
Cask crane support structure	I	a	-	-	-	(11)
Supports for Class I equipment	I	a,b	a,b,c	a,b,c	-	(7)
UHS Dam (Barrier Wall)	I	a	-	a	-	
SYSTEMS AND COMPONENTS						
A. Reactor Coolant System						
1. Reactor pressure vessel	I	b	c	b	A	
2. Reactor pressure vessel internals	I	b	c	b	-	
3. Control rod drive mechanisms	I	b	c	b	-	
4. Control element assemblies	I	b	c	b	-	
5. Pressurizer	I	b	c	b	A	
6. Steam generator						
a) Primary side	I	b	c	b	A	
b Secondary side	I	b	c	b	B	
7. Reactor coolant pumps	I	b	c	b	A	
8. Piping						
a Part of RCPB	I	b	c	b	A	(1)
9. Reactor protection instrumentation	I	b	c	b	-	(2)
B. Safety Injection System						
1 Safety Injection Tanks	I	-	-	-	B	(9)
2 Refueling Water Tank	I				B	(9)
3. Pumps	I	b	c	b	B	
4. Piping and valves						(3)
a) Part of RCPB	I	b	c	b	A	(1)
b) Required only for initial injection	I	b	c	b	B	
c) Required for long term post-accident cooling	I	b	c	b	B	
d) Normally isolated or automatically isolated from parts of system covered by (a), (b) or (c)	-	-	-	-	D	(4)
5. Instrumentation	I	b	c	b	-	(2)
C. Shutdown Cooling System						
1. Heat exchangers						
a) Reactor coolant side	I	b	c	b	B	
b) Component cooling water side	I	b	c	b	C	

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CLASS	TORNADO WIND CRITERION	FLOOD CRITERION	MISSILE CRITERION	QUALITY GROUP	NOTES
2. Piping and valves						(3)
a) Part of RCPB	I	b	c	b	A	(1)
b) Required for residual heat removal	I	b	c	b	B	
c) Normally isolated or automatically isolated from parts of system covered by (a) or (b)	-	-	-	-	D	(4)
3. Instrumentation	I	b	c	b	-	(2)
D. Chemical Volume and Control System						
1. Charging pumps	I	b	c	b	B	
2. Boric acid make-up tanks	I	b	c	b	B	
3. Boric acid pumps	I	b	c	b	B	
4. Letdown heat exchangers	-	-	-	-	B	
5. Regenerative heat exchanger	I	b	c	b	B	
6. Volume control tank	-	-	-	-	B	
7. Boric acid batching tank	-	-	-	-	D	
8. Ion exchangers	-	-	-	-	B	
9. Piping and valves						(3)
a) Part of RCPB	I	b	c	b	A	(1)
b) Required for letdown	-	-	-	-	C	(12)
c) Required for post-accident injection of boric acid	I	b	c	b	B	
d) Normally or automatically isolated from parts of system covered by (a), (b) or (c)	-	-	-	-	D	(4)
10. Instrumentation	I/non-I	b	c	b	-	(2)
E. Containment Spray System						
1. Pumps	I	-	-	-	B	
2. Nozzles	I	-	-	-	B	
3. Piping and valves						(3)
a) Required for spray and recirculation	I	-	-	-	B	
b) Normally or automatically isolated from parts of system covered by (a)	-	-	-	-	D	(4)
4. Instrumentation	I	-	-	-	-	(2)
F. Waste Management System						
1. Reactor coolant drain tank	-	-	-	-	D	
2. Flash tank	-	-	-	-	D	
3. Reactor drain pumps	-	-	-	-	D	
4. Holdup tanks	-	-	-	-	D	
5. Spent resin tank	-	-	-	-	D	
6. Flash tank pumps	-	-	-	-	D	
7. Gas surge tank	-	-	-	-	D	
8. Waste gas compressors	-	-	-	-	D	
9. Gas decay tanks	I	-	-	-	D	

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CLASS	TORNADO WIND CRITERION	FLOOD CRITERION	MISSILE CRITERION	QUALITY GROUP	NOTES
10. Piping and valves						(3)
a) For GDT Isol.	I	-	-	-	D	
b) For CNTMT Isol.	I	b	c	b	B	
c) Other	-	-	-	-	D	(3)(4)
11. Radiation monitoring instrumentation	-	b	c	b	-	(2)
G. Containment Cooling System						
1. Fan coolers	I	-	-	-	B	
2. Ductwork	I	-	-	-	B	
3. Instrumentation	I	-	-	-	-	(2)
H. Component Cooling System						
1. Pumps	I	a (8)	b	c (8)	C	
2. Surge tank	I	b	c	b	C	
3. Heat exchangers	I	a	b	c	C	
4. Piping and valves						(3)
a) Required for performance of safety functions	I	a	b	b,c	C	
b) Normally or automatically isolated from parts of system covered by (a)	-	-	-	-	D	(4)
5. Instrumentation	I	a	b	b,c	-	(2)
I. Cooling Water Systems						
1. Intake cooling water pumps	I	a (8)	b	c (8)	C	
2. Circulating water pumps	-	-	-	-	D	
3. Piping and valves						(3)
a) Required for performance of safety functions	I	a	b	b,c	C	
b) Normally or automatically isolated from parts of system covered by (a)	-	-	-	-	D	(4)
4. Instrumentation	I	a	b	c	-	(2)
J. Containment Isolation System						
1. Piping and valves (of all systems penetrating containment)						(3)
a) Part of RCPB	I	b	c	c	A	(1)
b) From first isolation valve inside containment or from containment penetration weld to outermost isolation valve (if not part of RCPB)	I	-	-	-	B	
2. Instrumentation	I	a,b	-	-	-	(2)

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CLASS	TORNADO WIND CRITERION	FLOOD CRITERION	MISSILE CRITERION	QUALITY GROUP	NOTES
K. Main Steam and Feedwater System						
1. Piping and Valves						(3)
a) From steam generator to outermost isolation valve	I	a	b	c	B	
b) Other	-	-	-	-	D	
2. Instrumentation	-	-	-	-	-	(2)
L. Auxiliary Feedwater System						
1. Pumps	I	a	b	b	C	
2. Condensate storage tank	I	a	a	b(10)	C	
3. Piping and valves						(3)
a) not normally or automatically isolated from Quality Group B components	I	a	b	c	B	(4)
b) Other	I	a	b	c	C	
4. Instrumentation	I	a	b	b	-	(2)
M. Emergency Power System						
1. Diesel generator sets	I	b	c	b	-	
2. Diesel oil storage tanks	I	c(8)	a	c(8)	C	
3. Diesel oil day tanks	I	b		b	C	
4. Diesel oil transfer pumps	I	a(8)	b	c(8)	C	
5. Diesel starting systems	I	b	c	b	C	
6. Diesel generator control boards	I	b	c	b	-	
7. Emergency switchgear	I	b	c	b	-	
8. Plant batteries and inverters	I	b	c	b	-	
9. Instrumentation	I	b	c	b	-	(2)
N. Sampling System						
1. Piping and valves						(3)
a) Part of RCPB	I	b	c	b	B	
b) Normally or automatically isolated from Quality Group A or B components	-	-	-	-	D	(4)
O. Hydrogen Control System	I	-	-	-	B	
I. Hydrogen recombiners	I	-	-	-	B	
2. Hydrogen purge system	I	-	-	-	C	(5)
3. Hydrogen sampling system	I	-	-	-	B	
P. Shield Building Ventilation System						
1. Fans	I	-	-	-	B	
2. Filters	I	-	-	-	-	
3. Ducting and dampers	I	-	-	-	B	
4. Instrumentation	I	-	-	-	-	(2)

TABLE 3.2-1 (Cont'd)

SYSTEMS AND COMPONENTS	SEISMIC CLASS	TORNADO WIND CRITERION	FLOOD CRITERION	MISSILE CRITERION	QUALITY GROUP	NOTES
Q. Ventilation System						(6)
1. Control Room AC and ventilation system						
- Indoor Sections	I	b	c	b	C	
- Outdoor Sections	I	-	b	-	C	
2. Engineered safety features area ventilation system	I	b	c	b	C	
3. Reactor auxiliary building main supply system	I	b	c	b	C	
4. Emergency switchgear ventilation system	I	b	c	b	C	

Footnotes:

Tornado wind and missile criteria

a = structure or component is designed to withstand design wind/tornado loadings and missile impacts.
b = components housed within a structure designed to withstand wind/tornado loadings and missile impacts.
c = separation of redundant components to preclude simultaneous failure by single missile impact.

Flood criteria

a = structures and components designed to withstand flooding effects.
b = positioning structures and components at sufficient elevation to preclude flooding.
c = components housed within waterproof structure.
d = cable is designed for operation in both wet or dry environments.

NOTES:

1. Refer to 10 CFR 50 Section 50.55 for definition of reactor coolant pressure boundary (RCPB). Components excluded by footnote 1(a) to Section 50.55a are Quality Group B.
2. Instrumentation required to actuate, maintain operation of, or detect failure of equipment needed to safely shutdown, isolate and maintain the reactor in a safe condition and prevent uncontrolled release of radioactivity to the environment is seismic Class I. Instrumentation designated as seismic Class I includes all sensing lines, except those whose breakage would cause the instrumentation to assume "fail safe" position. Non-seismic Class I sensing Lines from Class I piping or components shall be seismic Class I from the piping connection up to and including root valve.
3. Valves are of the same quality group as connected piping. Valves which comprise an interface between piping of different quality groups are of higher quality group.
4. Components of differing quality group other than Quality Group A may be considered to be normally isolated from each other if separated by at least one valve which is always closed during reactor operation or open during testing, sampling or other routine operation of short duration which is under administrative control. Such components may be considered to be automatically isolated if separated by a least one valve which closes automatically upon an appropriate engineering safety features actuation signal or by check valve which prevents flow from the higher to the lower quality group.
5. The only portion of the hydrogen purge system which is Quality Group B is that portion performing a containment isolation function.
6. Although this is not a steam or water containing system, it functionally corresponds to the Quality Group classification noted as per Safety Guide 26.
7. Supports are of the same safety class as the equipment that they support.
8. Protection for tornadic debris is to be provided on a backfit basis. (See Section 3.5.4.2 and Appendix 3F.)
9. Since the refueling water tank is not provided with missile shielding, the safety injection tanks have been credited as a backup water source for RCS makeup during safe shutdown (see Section 6.13 of Appendix 3F and Section 9.3.4.3.1).
10. The top of the Unit 1 condensate storage tank is not protected from vertical missiles; however, an intertie is provided to the fully missile protected Unit 2 condensate storage tank (see Section 6.10 of Appendix 3F and Section 10.5).
11. Cask crane support structure is designed for tornado wind only (not missile impact).
12. Quality Group B upstream of letdown control valves; Quality Group 'C' downstream to VCT outlet MOV.

TABLE 3.2-2

MINIMUM CODE REQUIREMENTS FOR QUALITY GROUPS

Component	Quality Group A	Quality Group B	Quality Group C	Quality Group D
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Class A	ASME Boiler and Pressure Vessel Code, Section III, Class C	ASME Boiler and Pressure Vessel Code, Section VIII Division 1	ASME Boiler and Pressure Vessel Code, Section VIII, Division I or Equivalent
Containment Vessel	-	ASME Boiler and Pressure Vessel Code, Section 111, Class B	-	-
0-15 Psig Storage Tanks	-	API-620	API-620	API-620 or Equivalent
Atmospheric Storage Tanks	-	Applicable Storage Tank Codes such as API-650, AWWAD100 or ANSI B 96.1	Applicable Storage Tank Codes such as API-650 AWWAD100 or ANSI B 96.1	API-650, AWWAD100 or ANSI B 96.1 or Equivalent
Piping[1]	ANSI B 31.7, Class I (1969 Edition)	ANSI B 31.7, Class II (1969 Edition)	ANSI B 31.7, Class III (1969 Edition)	ANSI B 31.1.0 or Equiva-lent (1967 Edition)
Pumps and Valves	Draft ASME Code for Pumps and Valves Class I	Draft ASME Code for Pumps and Valves Class II	Draft ASME Code for Pumps and Valves Class III	Valves - ANSI B 31.1.0 or Equivalent

Table 3.2-2 reflects minimum code requirements for Quality Groups used in original design. Replacement components may utilize alternate codes and edition/addenda as permitted by the PSL Unit 1 ASME Section XI program.

* Subsequent to the issuance of the ASME Code Section III all materials purchased for this service are qualified to ASME Section III.

[1] ANSI B31.7 was the Construction Code, however for piping, ASME BPV Code Section III, 1971 edition through Summer 1973 Addenda is used for Class II and Class III piping. ANSI B31.7 is still used for Class 1 pipe. Reconciliation was performed in accordance with ASME Section XI.

TABLE 3.2-2

MINIMUM CODE REQUIREMENTS FOR QUALITY GROUPS

Component	Quality Group A	Quality Group B	Quality Group C	Quality Group D
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Class A	ASME Boiler and Pressure Vessel Code, Section III, Class C	ASME Boiler and Pressure Vessel Code, Section VIII Division 1	ASME Boiler and Pressure Vessel Code, Section VIII, Division I or Equivalent
Containment Vessel	-	ASME Boiler and Pressure Vessel Code, Section 111, Class B	-	-
0-15 Psig Storage Tanks	-	API-620	API-620	API-620 or Equivalent
Atmospheric Storage Tanks	-	Applicable Storage Tank Codes such as API-650, AWWAD100 or ANSI B 96.1	Applicable Storage Tank Codes such as API-650 AWWAD100 or ANSI B 96.1	API-650, AWWAD100 or ANSI B 96.1 or Equivalent
Piping[1]	ANSI B 31.7, Class I (1969 Edition)	ANSI B 31.7, Class II (1969 Edition)	ANSI B 31.7, Class III (1969 Edition)	ANSI B 31.1.0 or Equivalent (1967 Edition)
Pumps and Valves	Draft ASME Code for Pumps and Valves Class I	Draft ASME Code for Pumps and Valves Class II	Draft ASME Code for Pumps and Valves Class III	Valves - ANSI B 31.1.0 or Equivalent

Table 3.2-2 reflects minimum code requirements for Quality Groups used in original design. Replacement components may utilize alternate codes and edition/addenda as permitted by the PSL Unit 1 ASME Section XI program.

* Subsequent to the issuance of the ASME Code Section III all materials purchased for this service are qualified to ASME Section III.

[1] ANSI B31.7 was the Construction Code, however for piping, ASME BPV Code Section III, 1971 edition through Summer 1973 Addenda is used for Class II and Class III piping. ANSI B31.7 is still used for Class 1 pipe. Reconciliation was performed in accordance with ASME Section XI.

3.3 WIND AND TORNADO LOADINGS

3.3.1 HURRICANE WIND CRITERIA

The design hurricane wind speed is 194 mph. The design wind speed was selected by reference to a PMH related to the site region with maximum sustained 30 foot over water winds of 140.6 mph, which was produced by a 158 mph maximum wind speed reduced by a near shore factor of 0.89. Assuming that such a wind would be associated with gusts which are 30 percent higher, the maximum gust would be 192 mph. The parameters which are used to arrive at the PMH are described in Section 2.4.5.1. Hurricane wind data and history are given in Section 2.3.2.2.

Storms of hurricane force (winds greater than 75 mph) cross the site approximately once in fifteen years. This figure is based on a hurricane frequency of one in ten years at West Palm Beach and one in twenty years at Vero Beach. The site is located about midway between these two cities.

Wind loads were determined and applied to all seismic Class I structures in accordance with procedures incorporated in Reference I based on the design wind of 194 mph. In no case did the 194 mph wind govern the design of structures.

3.3.2 TORNADO CRITERIA

Structures or components whose failures could prevent safe shutdown of the reactor or result in significant uncontrolled release of radioactivity to the environment, are protected from such failure due to tornadic wind loading and associated differential pressure by:

a) design of structures or components to withstand such wind loading

b) locating components within structures designed to withstand such wind loading

Table 3.2-1 lists tornadic wind protection classifications for plant structures, systems and components. The a or b designation in the table refers to a or b above.

The design tornado has a rotational wind speed of 300 mph and a translational speed of 60 mph. Since the widths of the major Class I structures (the shield building and reactor auxiliary building) are relatively larger than the distribution over which the combined effects of the rotational and translational velocities are postulated to act, a 300 mph wind speed was used for the reactor building and reactor auxiliary building.

The design tornado applied to this site is extremely conservative. Florida tornadoes are much less severe. See Section 2.3.1.3.

The diesel generator building is designed for 300 mph wind speed on the basis of its low height. The fuel handling building is designed for the full 360 mph wind speed because this structure is too narrow and too low to meet the 300 mph wind speed criteria established for other Class I structures.

The shield building has a radius of 154 feet while the overall plan dimensions of the reactor auxiliary building are approximately 115 by 240 feet. Data extrapolated from Reference 2 indicate an average band width of approximately 50 to 80 feet over which the combined velocity distribution of 360 mph is postulated to act. On this basis, a uniform wind speed of 300 mph for large Class I structures was adopted for design of the shield building and reactor auxiliary building. In addition to the effect of the design wind speed associated with this tornado, a 3 psi pressure differential in 3 seconds is applied simultaneously with the wind loading to the seismic Class I structures except for diesel generator building. Because of the ventilation openings in the diesel generator building, the structure is designed for a 2.25 psi pressure differential.

The tornado wind speed is converted into equivalent static pressure loading and the computations for wind pressure, their distribution on surface area of buildings, shape factors and drag coefficient are based on the procedures outlined in Reference 1. Because of the unique characteristics of tornadoes, gust factor and velocity variation with height are not considered. With respect to the pressure distribution around the dome-cylinder shield structure wind force data reported in Reference 3 was used in the design. Equivalent static pressure loading for the various structures are given on Table 3.3-1.

The turbine building is the only structure not designed for tornado wind which can be considered in proximity to safety related equipment and structures. Under tornado loading, the first failures anticipated in the turbine building would occur in the vertical bracing system. The buckling of some of these members would force some beam-to-column connections to utilize their inherent moment-resisting capability and behave as moment connections. This in turn would result in the local overstress of some connections. As they begin to yield, the load would redistribute itself among other parts of the frame and the structure would behave in a plastic manner.

The failures that would occur under these conditions are anticipated as being of a local nature. It is not credible that the building will collapse because the turbine pedestal will also act to restrain the structure. In summary, it is not anticipated that any structure or equipment necessary for safe plant shutdown would be affected by local failures in the turbine building due to tornado loadings.

Tornado generated missiles considered in the plant design include a 10ft long 2 inch x 4 inch timber traveling at 360 mph or a 4000 lb automobile traveling at 50 mph. The analysis to determine the effect of missiles is described in Section 3.5.3. Missile loadings are not applied simultaneously with wind loadings. The purpose of the tornado missile analysis is to determine that the structure can absorb sufficient energy to completely stop the missile without penetration. The capability of the facility to accommodate tornado generated missiles is discussed in Appendix 3F.

REFERENCES FOR SECTION 3.3

1. ASCE Paper No. 3269, "Wind Forces on Structures," 1961.

2. Hoecker, W. H., Jr., "Three Dimensional Pressure Pattern of the Dallas Tornado and Some Resultant Implications," Monthly Weather Review, December 1961.

3. Maher, F. J., "Wind Loads on Dome-Cylinder and Dome-Cone Shapes," Proceeding, ASCE, Volume 92, No. 575, Paper 4933, October 1966.

TABLE 3.3-1

TORNADO WIND SPEEDS AND

RESULTING STATIC PRESSURE LOADINGS

Structure	Tornado Wind Speed (mph)	Gust Factor	External Pressure Coefficient	External Loading (Psf)	External Loading with Pressure Diff.
Reactor Building	300	1		See Figures 3.8-35 to 3.8-38	
Reactor Aux. Building	300	1	.9	217 (1)	220 (1)
			.5	115 (2)	550 (2)
			.5	115 (3)	270 (3)
Fuel Handling Building	360	1	.9	300 (1)	240 (1)
			.5	166 (2)	706 (2)
			.8	266 (3)	525 (3)
Diesel Generator Building	300	1	.9	217 (1)	110 (1)
			.5	115 (2)	440 (2)
			.5	115 (3)	230 (3)

(1) Windward
(2) Leeward
(3) Roof (includes external wind pressure, internal pressure differential and slab dead weight)

3.4 WATER LEVEL (FLOOD) DESIGN

3.4.1 FLOOD ELEVATIONS

The plant area is situated above the highest possible water levels attainable except for wave runup resulting from probable maximum hurricane (PMH) considerations. During this condition, wave runups to 17.2 ft* mean low water (MLW) are possible. Plant grade is at elevation 18 ft and minimum entrance elevation to all safety related buildings is +19.5 ft. The maximum elevation of roadways on the plant site is +19.0 ft., thus any ponding of water that might result will be below the building entrances.

3.4.2 PHENOMENA CONSIDERED IN DESIGN LOAD CALCULATIONS

All seismic class I structures are designed to withstand buoyant and static forces associated with high water levels. Only minimum structure and equipment deadweight are used in these calculations.

3.4.3 FLOOD FORCE APPLICATION

In essence, structural components of all seismic Class I structures are subjected to a buoyant soil loading condition up to elevation +16.2 ft and a saturated soil loading condition from elevation + 16.2 ft to grade. This condition is the buoyant loading condition and accounts for conditions of maximum buoyance and flooding.

3.4.4 FLOOD PROTECTION

Structures and components whose failure could prevent safe shutdown of the plant or result in significant uncontrolled release of radioactivity are protected from the effects of high water levels and wave runup associated with PMH conditions by one or more of the following:

a) Design of structures and components to withstand such effects where functionally required.

b) Positioning of the structures and components such that they are located at sufficient grade to preclude inoperability due to external flooding.

c) Housing within waterproof structures: The shield building and reactor auxiliary building are the only seismic Class I structures with basements. These structures are completely waterproofed to finish grade with Nob-Lok waterproofing. All construction joints are waterstooped with 6 in. polyvinyl chloride.

Table 3.2-1 lists the flood protection criteria applied to plant structures, systems and components. The a, b, or c designation in the table refers to items a, b, or c above.

The list below designates each seismic Class I structure and the associated means of protection from flooding.

* Reference Section 2.4.5.9 for updated surge levels and wave runup analysis.

Structure	Flood Protection
Shield Building	No openings below elevation +22 ft
Reactor Auxiliary Building	Ground level openings at 19.5 ft
Fuel Handling Building	Ground level openings at 19.5 ft
Diesel Generator Building	Floor and equipment above elevation +22 ft
Intake Structure	Motors located above elevation +22 ft

All buildings with the exception of the turbine building are of the enclosed building type. The turbine building will be subjected to wind driven water spray, consequently, all equipment inside this building is designed for outdoor service.

The flood protection for the emergency diesel generator system includes protection for the oil storage tanks which rest at elevation 22 ft. The outlet nozzle for pump suction is located at elevation 22 ft 6 in., with filling connections at approximately 37 ft and vent connections at approximately 38 ft.

All permanent door openings in the exterior walls of the reactor auxiliary, fuel handling and diesel generator buildings are provided with either roll-up or swing type doors for protection from rain, wind and other atmospheric effects. The rolling shutter doors are fabricated of interlocking slats, curved and jointed to shed water. Large doors are furnished with a continuous, adjustable rubber stripping at jambs, head and floor to provide a positive weather-tight closure. Access doors may not be provided with weather-stripping in all cases, however, the amount of leakage-induced flooding through these doors is not more adverse than that considered in the analysis presented in Section 3.1.3 of Chapter 9.5A on the rupture of non-seismic Class I equipment (fire system piping).

Waterproofing details of penetrations below El + 22.0 feet MLW for safety related buildings are shown in Figure 3.4-2. All external building penetrations are waterproofed and/or flood protected to preclude the failure of a safety related system or component due to external flooding. All penetrations for pipes or electrical ducts are either encased in concrete where they penetrate the wall, or, when sleeves are used, enclosed in a pipe boot designed to prevent seepage. The end result is a completely waterproofed structure below grade. Boots are not used below the normal ground water table.

All interconnections between safety related structures that could be subjected to flooding are waterproofed as indicated in Figure 3.4-2.

As demonstrated in Sections 2.4.5.6 and 2.4.5.7, the need for additional flood protection beyond what is provided by the elevations of the openings of the safety related structures is not required to protect any of the safety related structures from wave runup or wind driven rain, even during a probable maximum hurricane. Therefore, the use of gasketed aluminum stop logs and/or sandbags and plastic sheeting is not required.

PAGE 3.4-5 INTENTIONALLY BLANK

The site drainage system is designed to preclude flooding of safety related structures and components under PMH conditions however total flooding of the drain lines will not cause water to backup into areas which would jeopardize the required function of a safety related system. PMH conditions would produce an overland flow which would exceed the capacity of the drain lines, however, excess waters would run off the plant island. Section 2.4.2.3 addresses drainage of water from the southern site property and the effect of water pooling caused by the intake canal berm.

The Unit 1 and Unit 2 site drainage plans are shown in Figures 3.4-3 and 3.4-4, respectively. In areas where Unit 1 drain lines are to carry storm water from both units, the lines are oversized to accommodate the additional flow. The interfaces between Unit 1 and Unit 2 drainage lines are shown in Figure 3.4-3.

Drain lines are sized to accommodate runoff in the plant area. Runoff in the plant area is estimated by relating the tributary area and the rainfall intensity to an estimated proportion of the rainfall reaching the catch basin as direct runoff.

This procedure is represented by the following formula:

$$Q = ACI_P$$

where Q = design discharge, cfs
A = tributary drainage area, sq ft
C = runoff coefficient based on surface conditions
I = intensity of rainfall, in/hr
P = coefficient based on percent of full pipe flow

The design considered values of C consistent with use in the Rational Method for various ground surface types. The intensity of rainfall, I, used in the calculations was 6 inches per hour. The tributary drainage area was determined by the location of surrounding catch basins and storm drain lines.

The ISFSI drainage plan is shown in Figure 3.4-5.

Catch basins are constructed to provide ready access to storm drains for inspection and maintenance as well as to serve as points of concentration for runoff. Runoff computations for catch basins include roof, floor and equipment drains having no potential for contamination.

The analyses of a postulated failure of a pressurized fire main within the reactor auxiliary building is provided in Subsection 3.1.3 of Chapter 9.5A. This is the only plant structure that houses both pressurized fire system piping and safety related equipment. The consequences of this postulated failure are acceptable.

FIGURE 3.4-1

HAS BEEN INTENTIONALLY DELETED

Refer to drawing
8770-G-584

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR AUXILIARY BUILDING EXT. WALLS-
MISC DETAILS-M&R

FIGURE 3.4-2

Refer to drawing
8770-G-483

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

GRADING & DRAINAGE – SH.1

FIGURE 3.4-3

Refer to drawing
2998-G-483

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

SITE GRADING & DRAINAGE-SH.1

FIGURE 3.4-4

Refer to drawing
8770-G-921 SH 9

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

ISFSI PROJECT - GRADING & DRAINAGE
DRAINAGE PLAN

FIGURE 3.4-5

3.5 MISSILE PROTECTION

This section discusses the missile protection criteria, potential missile sources and methods of missile protection for safety related structures and equipment.

The unit is designed so that missiles from internal sources do not:

a) cause or increase the severity of a loss of coolant accident (LOCA)

b) damage engineered safety features when their operation is required to mitigate the consequences of an accident

c) prevent safe shutdown and isolation of the reactor

d) jeopardize primary containment function as a radioactive material barrier during and following accidents that release radioactive material into the containment vessel

The unit is designed so that missiles from external sources do not:

a) cause a LOCA

b) damage fuel stored in the spent fuel pool

c) prevent safe shutdown and isolation of the reactor

d) damage systems or components whose failure could result in significant uncontrolled release of radioactivity

e) jeopardize structural integrity of seismic Class I structures

3.5.1 MISSILE BARRIERS AND LOADINGS

Wherever possible, component and system design preclude the generation of missiles. This is achieved by suitable choice of materials, normal and faulted stress levels, and system and component characteristics which avoid missile producing effects, even under faulted conditions.

Systems and components which are identified as potential missile sources are, wherever possible, arranged and oriented such that the target structure or component is capable of withstanding the impact and the design criteria are not violated.

Barriers are provided for missiles which cannot be positioned to take advantage of existing structures and which could cause failure of safety related structures or components. Generally these barriers are designed to contain or to deflect the missiles from the safety related feature. As a minimum requirement, penetration of the missile through the barrier will reduce the missile energy to levels which cannot compromise the safety feature function.

Wherever possible, advantage is taken of walls and structures arising from functional requirements, other than missile considerations, by judicious arrangement of equipment.

3.5.2 MISSILE SELECTION

3.5.2.1 Internal Missiles

a) Reactor Building

Internal missiles which could be generated from pressure containing components that are part of the reactor coolant system or main steam system are considered in the design of the reactor building. The entire reactor coolant system and parts of the main steam system are surrounded by the secondary shield wall and their components arranged so that a missile generated from one component will not damage its counterparts. In general, the secondary shield wall protects the containment vessel from missiles generated within the secondary shield. Other shields include the primary shield wall which surrounds the reactor vessel and the top shield which is located above the control element drive mechanism.

The procedures used to calculate the kinetic energies of potential missiles inside the containment were taken from reference (1). The values are based on the following conservative assumptions:

1) The effect of gravity is ignored

2) All the potential energy of fasteners is converted into kinetic energy

3) Missiles propelled by expanding fluid are accelerated for an additional distance

4) Reactor coolant expands as a two phase fluid

b) Reactor Auxiliary Building

The only area in the reactor auxiliary building which contains both high pressure equipment and safety related equipment is the emergency core cooling pump room located at the -10 ft elevation. The -10 ft elevation of the reactor auxiliary building is divided into two compartments by a combination missile shield/flood wall. Each compartment contains the minimum complement of safety injection and containment spray pumps necessary to mitigate the effects of a LOCA.

c) Diesel Generator Building

The potential internal missile considered in the design of the diesel generator building is a ruptured compressed air starting tank. There are four compressed air tanks per diesel generator unit each with a width of 30 in., height of approximately 7 ft and operating pressure of 200 psi.

Each of the four tanks in each set is bolted to a common skid by four 1/2 in. diameter bolts. The skids are bolted to the diesel generator building floor. All of this equipment has been designed to seismic Class I specifications. Furthermore, each of the four tanks are connected to each other by two 2 in. diameter horizontal pipes (used for pressurizing the tanks) which would aid in dissipating the potential missile energy.

Disregarding all of the above and assuming a tank could become a missile, the shield wall which separates the diesel generator building into two redundant parts is sufficient to withstand the highest impact force which could be imparted by such a ruptured tank.

3.5.2.2 External Missiles

a) Tornado Missiles

Impact by missiles which could be generated by tornado winds are considered in the design of safety related structures and components. The postulated missiles considered include representative objects which could be picked up in the plant area and propelled by the tornado winds, such as automobiles and wooden planks. (See Appendix 3F.)

b) Turbine Missiles

The analysis of turbine missile probabilities is discussed in Section 3.5.3.2.

3.5.3 SELECTED MISSILES

3.5.3.1 Internal Missiles

Table 3.5-1 lists the spectrum of potential internal missiles, their kinetic energy, weights, leading cross-section configurations and the barriers designed to withstand them. The basic formula used to calculate missile penetration is the modified Petry formula, as derived in reference (2).

An analysis of the potential missiles listed in Table 3.5-1 indicates that a large number of potential missiles have sufficiently low kinetic energy to be of any consequence. All items discussed below refer to Table 3.5-1:

a) Items a-1, 2, 3 and 4 are directed toward the missile shield but have insufficient energies to penetrate

b) All potential missiles in item c are directed toward the pressurizer shield cavity which is designed to accept the load

c) Item d, control rod drive assembly, is directed toward the secondary shield wall which is designed to accept the load

d) The remaining items of Table 3.5-1 are of insufficient energies to damage vital equipment. These items include the original steam generator manway and handhole studs and nuts. A calculation of the kinetic energy of the Replacement Steam Generator (RSG) primary manway stud and nut indicates that the RSG manway and handhole studs and nuts would also be low energy missiles.

The main steam and feedwater piping are analyzed considering the impacts of potential missiles (g) and (h) of Table 3.5-1.

3.5.3.2 <u>External Missiles</u>

a) Tornado Missiles

Tornado generated missiles considered in the design include the following:

Missile	Velocity	Energy
2" x 4" x 10' wooden plank	360 mph	1.33×10^5 ft-lbs
4000 lb automobile	50 mph	3.35×10^5 ft-lbs

The external walls of the reactor building, reactor auxiliary building, fuel handling building and diesel generator building are designed to withstand the tornado generated missiles. The intake structure and associated valve pit are designed to protect safety related cooling water piping from tornado generated missiles. The equation used to evaluate the missile penetration is as follows (Reference 2):

$$D = K\,A_P\,\log_{10}\left[1 + \frac{V^2}{215{,}000}\right]$$

where:

D = depth of penetration (ft)

K = coefficient of penetration

A_p = missile wt ÷ cross-sectional area

V = missile velocity (ft/sec)

The required concrete wall thickness is 2 x D.

Additional tornado missiles have been analyzed for their effect on plant structures and equipment using the characteristics presented in Table 3.5-1A. Using the above formula for penetration, it is concluded that the plant concrete walls are capable of resisting any penetration by these postulated missiles.

Below is a listing of the calculational assumptions used and a description of the conservatism contained therein.

Assumption	Degree of Conservatism
A) Effective Drag Coefficient: $C_e = C_d A_e/W$	A) An effective drag factor (C_e) is used based on an effective area (A_e). The effective area, A_e, is assumed instead of the orientation of any of the maximum area because the orientation of any of the missiles will change with relation to the wind, thus resulting in a lower velocity than that obtained using the maximum area. The turbulent cyclonic tornado winds are not conducive to missiles with a fixed orientation with respect to the wind direction. Rather, tumbling or a continuing reorientation of the missile with regard to wind is expected.
B) Area of Impact	B) The minimum area of impact is used in the analysis resulting in the largest impact penetration. This is the most conservative approach.
C) Height of Impact	C) Although some missiles are indicated as not rising more than a certain distance above grade, all missiles with the exception of the utility pole and the auto are applied to all, Class I structures regardless of height. For these missiles this represents the most conservative approach.

Per the Staffs request (Q2.25 May 15, 1974 Staff letter), another analysis has been conducted assuming a fixed orientation of the missile such that $C=C_dA_{max}/W$. This analysis is conducted to assess the upper limit of penetrability of missiles only. It is not considered to be a design basis analysis. The analysis also assumes a minimum impact area; in other words, it is assumed that suddenly the missile reorients itself to strike in the most penetrating position. The resultant elevations and velocities are presented in Table 3.5-1B.

The upper limit analysis for penetrability of structures indicates that for all the missile heights achieved (Table 3.5-1B) total penetration of the structures will not occur.

Pursuant to a Staff request of August 26, 1974, the capability of the site to accommodate tornado missiles has been evaluated in detail. This analysis is provided as Appendix 3F. It reaffirms the Unit 1 design basis for tornado missiles, and indicates where enhancement of capability to accommodate tornadic debris (spectrum of light objects) could be achieved. (See also Section 3.5.4.2).

b) Turbine Missiles

Modern manufacturing and quality control procedures have eliminated the credibility of turbine rotor failures. To ensure this, FPL complies with the turbine vendor's inspection/refurbishment recommendations.

The main turbine is a Siemens Energy, Inc. unit consisting of one high pressure (HP) and two low pressure (LP) elements as shown on Figures 3.5-1 through 3.5-3.

For many years, Siemens Energy, Inc. original design of shrunk on disk rotors, as well as the advanced disk design, have demonstrated and proven the quality of this technology. The total number of fleet operating hours is more than 2,750,000 which have led to more than 40,000,000 disk operating hours, bearing in mind that each unit consists of two to three LP turbine elements with six to ten disks each. The oldest rotors have been in operation for approximately 225,000 operating hours, and the inspections of the disks performed after more than 200,000 hours detected no cracks.

Several important factors have been contributed to this record:

1) Factory test procedures

Destructive testing of material specimens taken from the disc forgings and ultrasonic test of each disc following major heat treatment ensure sound discs with mechanical properties (tensile strength, yield strength, ductility and impact strength) equal to or exceeding the specified levels.

2) Redundancy in the control system

The turbine generator is provided with three overspeed protection systems, overspeed protection controller (OPC) and two redundant electronic overspeed protection systems. The OPC (electro-hydraulic) control system and the primary electronic overspeed protection system do not share any sensing devices. These are discussed in detail in Section 10.2.2.

On a unit trip, two separate main steam line valves (stop and governing valves) are tripped closed to provide a redundant system.

It should be noted that each stop, governing, reheat stop and intercept valve is spring-closed; thus, it is only necessary to dump the high pressure fluid under the servo-actuators to close the valves.

3) Operating test procedures

Routine testing of the turbine steam inlet valves and the emergency overspeed protective system serve to verify continued operability of the overspeed protection.

4) High pressure turbine construction and design

The high pressure turbine element, as shown in Figure 3.5-1, is of a double flow design thus it is inherently thrust-balanced. Steam from the four control valves enters at the center of the turbine element through four inlet pipes, two in the base and two in the cover. These pipes feed four double-flow nozzle chambers flexibly connected to the turbine casing. Steam leaving the nozzle chambers passes through the diagonal stage and flows through four reaction stages, all mounted on the inner casing upstream of the extraction. Downstream of the extraction, steam flows through four reaction stages mounted on the guide blade carriers, shown in Figure 3.5-2. The inner casing and the guide blade carriers are mounted on the outer casing.

The high pressure rotor is made of 26NiCrMoV10-10 alloy steel. The rotating blades are made of X20Cr13 high chromium steel. The rotor with rotating blades weighs approximately 121,916 lb.

The inner casing and the guide blade carriers are GX8CrNi12 high chromium steel castings. The diagonal stage guide blades are made of X22CrMoV12-1 high chromium steel. The reaction stage guide blades are made of X20Cr13 high chromium steel. The inner casing with stationary blades weighs approximately 48,700 lb., and each guide blade carrier with stationary blades weighs approximately 27,000 lb.

The outer casing cover and base (upper and lower half) are tied together by means of more than 100 studs. The horizontal joint plane studs end cap nuts and the keys and support plates are made of X19CrMoNbVN11-1 high chromium steel.

Studs have lengths ranging from 17 to 66 inches and diameters ranging from 2.75 inches to 4.5 inches.

All fragments generated by any postulated failure of the HP turbine rotor would be contained by the HP turbine inner casing, guide blade carriers, and outer casing.

5) Low pressure turbine construction and design

The double flow low pressure turbine, shown in Figure 3.5-3, incorporates high efficiency blading, diffuser type exhaust and liberal exhaust hood design. The low pressure turbine casing is fabricated from steel plate to provide uniform wall thickness, reducing thermal distortion to a minimum. The entire outer casing is subjected to low temperature exhaust steam.

The temperature drop of the steam from its inlet to the LP turbine to its exhaust from the last rotating blades is taken across three walls; a guide blade carrier, a thermal shield, and an inner casing as shown in Figure 3.5-3. This precludes a large temperature drop across any one wall, except the thermal shield which is not a structural element, thereby virtually eliminating thermal distortion. The fabricated inner casing is supported by the outer casing at the horizontal centerline and is fixed transversely at the top and bottom and axially at the centerline of the steam inlets, thus allowing freedom of expansion independent of the outer casing. The guide blade carrier is, in turn, supported by inner casing at the horizontal centerline and fixed transversely at the top and bottom and axially at the centerline of the steam inlets, thus allowing freedom of expansion independent of inner casing. The inner casing is surrounded by the thermal shield. The steam leaving the last row of blades flows into the diffuser where the velocity energy is converted to pressure energy.

The outer casing is fabricated mainly of ASTM 515-GR65 material. The inner casing is fabricated of ASTM A 516-GR60 and the guide blade carrier is fabricated of ASTM A 508 material.

The low pressure rotors are made of NiCrMoV alloy steel.

The discs are made of NiCrMoV alloy steel. There are six discs shrunk on the shaft with three per flow. These discs experience different degrees of stress when in operation. Disc No. 1, starting from the transverse centerline, experiences the highest stress, while Disc No. 3 experiences the lowest.

6) Probability of postulated failures of LP turbine discs

In view of operating experience and NRC safety objectives, the NRC staff has shifted emphasis in the reviews of the turbine missile issue from the strike and damage probability (P_2xP_3) to the missile generation probability (P_1) and, in the process, has attempted to integrate the various aspects of the issue into a single, coherent evaluation.

Through experience of reviewing various licensing applications, the staff has concluded that P_2xP_3 analyses provide only “ball park” or “order of magnitude” values. Based on simple estimates for a variety of plant layouts, the staff also concludes that the strike and damage probability product (P_2xP_3) can be reasonably taken to fall in a characteristic narrow range which is dependent on the gross features of plant layout with respect to turbine generator orientation; i.e., (a) for favorable oriented turbine generators P_2xP_3 tends to lie in the range of 10^{-3} to 10^{-2}. In addition, detailed analyses such as those discussed in this evaluation show that, depending on the specific combination of material properties, operating environment, and maintenance practices, P_1 can have values from 10^{-9} to 10^{-1} per turbine year depending on the turbine test and inspection intervals. For these reasons, in the evaluation of $P_4 = (P_1xP_2xP_3)$ the probability of unacceptable damage to safety-related systems from potential turbine missile, the staff is giving credit for the product of the strike and damage probabilities (P_2xP_3) of 10^{-3} for a favorably oriented turbine and 10^{-2} for an unfavorably oriented turbine (St. Lucie orientation), and is discouraging the elaborate calculation of these values.

By maintaining an initial small value of P_1 through turbine testing and inspection provides a reliable means of ensuring that the objectives precluding turbine missiles and unacceptable damage to safety-related structures, systems, and components can be met. It simplifies and improves procedures for evaluation of turbine missile risks and ensures that the public health and safety is maintained.

For these reasons, strike and damage calculations were not performed for the St. Lucie Unit 1 replacement low pressure turbine. A methodology has been developed by Siemens Energy Inc. and approved by NRC (Ref. 6). This methodology determines the probability of an external missile (P_1) to be the sum of the probability of an external missile for turbine speeds up to 120% of rated speed (P_r) and the probability of an external missile for turbine speeds greater than 120% of rated speed (P_0). This methodology determines the external missile probability based on a turbine disc inspection interval of 100,000 hrs and quarterly turbine valve tests provided that no cracks are detected in the discs. These results are then compared to the NRC minimum reliability requirement of $P_1<10^{-4}$/yr for favorably oriented turbines and $P_1< 10^{-5}$/yr for unfavorably oriented turbines (St. Lucie orientation) (Ref. 5). In order to apply the approved methodology, the NRC requires the following:

a. The approximate date for the turbine disc inspection at the end of 100,000 hrs of operation of the rotors,

b. A commitment to inform the NRC about the turbine disc inspection results and plans to reduce the probability of turbine missile generation, P_1, for continued operation should cracks be detected in the inspection, and

c. Justification for any additional turbine missile analyses, or minor deviations that may be plant specific.

A missile probability analysis was then performed for the St. Lucie Unit 1 low pressure turbines which include the upgraded BB281-13.9m2 rotors with Advanced Disc Design shrunk-on discs (Ref. 7) by applying the currently approved methodology (Ref. 6) along with the extended 6 month valve test interval. Based on the conservative assumptions applied, the probability of an external missile for speeds up to 120% of rated speed (P_r) is $2.88x10^{-7}$/yr for a disc inspection interval of 100,000 operating hours. Applying the 6 month valve test interval, the probability of an external missile for speeds greater than 120% of rated speed (P_0) is $1.59x10^{-6}$/yr for a disc inspection interval of 100,000 operating hours. Therefore, $P_1 = P_r + P_0 = 1.88 \times 10^{-6}$/yr which can be compared to the NRC limit of $1.0x10^{-5}$/yr (i.e., $11.42x10^{-5}$/yr for a disc inspection interval of 100,000 operating hours) to demonstrate the probability of an external missile is well below the NRC limit and the Unit can be operated for 100,000 hrs between disc inspections provided no cracking is detected.

The turbine disc inspection interval will begin 100,000 operating hours following the PSL 1-24 refueling outage. The NRC will be informed of turbine disc inspection results and plans to reduce the probability of turbine missile generation P_1, for continued operation should cracks be detected during the inspection.

(text deleted)

3.5.4 BARRIER DESIGN PROCEDURES

The analysis and design for the missile loads are based upon the elastic impact between missile and missile shield. Energy loss during impact is calculated in accordance with the "Structural Design for Dynamic Loads," by C. H. Norris and others. Local penetration is calculated in accordance with the "Design for Protective Structures," by Arsham Amirikian, Department of the Navy. The allowable stress increase due to the instantaneous behavior of missile load is in accordance with the "Design of Structures to Resist Nuclear Weapon Effects," ASCE Manual No. 42.

The design criteria and additional protection provided to enhance the facility's tornado resistance capability is addressed separately in Appendix 3F.

3.5.4.1 Internal Missiles

a) Reactor Building

The effects of missile generation are limited by surrounding potential missile sources with a missile barrier designed to contain the missile within a specific area.

b) Reactor Auxiliary Building

The -10 feet elevation of the reactor auxiliary building is divided into two compartments by a combination missile shield/flood wall. This is the only area of the reactor auxiliary building that contains both high pressure equipment and safety related equipment.

c) Diesel Generator Building

The common wall and connecting door separating the units is designed to withstand missiles developed on either side.

3.5.4.2 External Missiles

a) Tornado Missiles

Plant structures, systems and components required for safe shutdown are protected from the effects of a tornado missile by any of the following means:

1) Design of structures, systems or components to withstand missile impact

2) Protection of systems or components by structures designed to withstand missile impact

3) Separation of redundant components to preclude simultaneous failure by single missile impact

Table 3.2-1 lists the missile protection criteria applied to all safety related structures, systems and components. The a, b, or c designation refers to items 1, 2 and 3 above.

Redundant exposed and underground outdoor equipment and piping required for safe shutdown such as the auxiliary feedwater pumps, component cooling water pumps, and intake cooling water pumps are separated according to the criteria set forth in Section 9.2.1.3.4 or protected in accordance with Appendix 3F. Underground cabling is provided with a minimum of 3 ft. of soil cover including a 9 or 15-inch reinforced concrete protective slab. In areas where redundant under ground cables must cross paths, one of the cables is buried to a depth of from 4 ft. to 5 ft. with a concrete slab placed between the cables. All other components required for safe shutdown are located within or protected by structures designed to withstand tornado missile impact such as : reactor building, reactor auxiliary building, diesel generator building and the condensate storage tank enclosure. In addition, the fuel handling building and the main steam and feedwater trestles are designed to withstand a tornado missile.

Appendix 3F provides an evaluation of the tornado design. It analyzes the capability to accommodate tornado missiles across the spectrum of possible missiles from the numerous light objects that will undoubtedly be found within the tornado wind field (tornadic debris) to the large high energy design basis type missiles, i.e., the 2 X 4 plank and the automobile. The evaluation concludes that:

(a) Based on meteorological considerations, supported by 85 years of historic data, the tornadic risk associated with the intense design basis type tornado is considerably less in peninsular Florida than in areas to the north and west.

(a) The use of outdoor separated redundant components results in an acceptably low probability for loss of redundant components concurrent with a tornado strike at the site.

In addition the review indicated that enhancement of the facility's tornado resistance capability can be achieved by providing additional protection from tornadic debris. Specifically this involves the following active components and structures:

(1) ICW pumps

(2) CCW pumps

(3) Diesel Generator Building openings

(4) Diesel oil transfer pumps

(5) Interconnecting the Unit 1 and Unit 2, (future) D. O. storage tanks.

(6) Auxiliary Feed Pumps and inteconnection the Unit 1 and Unit 2 Condensate Storage Tanks.

Protection for tornadic debris in these areas, is provided commensurate with the Commission's guidance on backfit as provided by 10 CFR 50.109. The design of the protection satisfies the following criteria:

(1) The protection does not provide an additional source of tornadic missiles, i.e., it accommodates winds associated with the Staff's 360 mph tornado.

(2) The protection accommodates a seismic event (DBE) without loss of function, or its failure does not adversely affect a safety related component or structure.

(3) The protection is compatible with existing plant structures, maintenance requirements, and component operability requirements.

b) Turbine Missiles

Based on the probability analysis results described in Section 3.5.3.2, there are no turbine missiles being postulated.

3.5.5 MISSILE BARRIER FEATURES

The plant general arrangement (Figures 1.2-1 through 1.2-19) shows the layout of structures used as missile barriers. Refer to Figure 3.5-6 for details of the reactor missile shield.

REFERENCES FOR SECTION 3.5

1. R. C. Gwaltney, "Missile Generation and Protection in Light Water Reactors," ORNL-NSIC-22, Oak Ridge National Laboratory, March 1, 1967.

2. Arsham Amirikian, "Design of Protective Structures," Report NP 3726, Bureau of Yards and Docks, Department of the Navy, August 1950.

3. Nuclear Regulatory Commission, Regulatory Guide 1.115, Rev. 1, “Protection Against Low Trajectory Turbine Missiles”, July 1977.

4. “Standard Review Plan, Section 3.5.1.3 – Turbine Missiles”, NUREG-75/087, Revision 1- Nuclear Regulatory Commission.

5. NUREG-1048, Appendix U, Supplement No. 6, “Safety Evaluation Report Related to the Operation of Hope Creek Generating Station”

6. TP-04124-NP-A, “Missile Probability Analysis for the Siemens 13.9 m^2 Retrofit Design of Low-Pressure Turbine by Siemens AG”, June 7, 2004 (includes NRC Final Safety Evaluation letter dated March 30, 2004)

7. CT-27344, Rev. 1 “Missile Report for FPL St. Lucie Units 1&2, BB281-13.9 m^2”, August 31, 2009, Siemens Energy, Inc.

TABLE 3.5-1

INTERNAL MISSILE PARAMETERS

Item	Kinetic Energy (Ft-Lb)	Weight (Lb)	Leading Section	Structure/Shield/Barrier
a) Reactor Vessel				
1) Closure Head Nut	2,022	116	Annular Ring, OD=10-9/16", ID=6.8"	Missile Shield on Reactor Vessel
2) Closure Head Nut & Stud	4,932	710	Solid Circle 7" in Diameter	Missile Shield on Reactor Vessel
3) Instrumentation Assembly	127,000	335	Solid Disk 6-1/2" Diameter and 3" Thick	Missile Shield on Reactor Vessel
4) Instrumentation from Flange Up	144,000	165	Solid Disk 6-1/2" Diameter and 3" Thick	Missile Shield on Reactor Vessel
5) Instrument Flange Stud	14.3	6-1/2	Solid Circle 1-1/2" Diameter	Missile Shield on Reactor Vessel
b) Steam Generator				
1) Primary Manway Stud and Nut	71	4-1/4	Solid Circle 1-1/2" Diameter	Low Energy
2) Secondary Handhole Stud and Nut	8	1-3/4	Solid Circle 1" Diameter	Low Energy
3) Secondary Manway Stud	38	4.6	Solid Circle 1-1/4" Diameter	Low Energy

TABLE 3.5-1 (Cont'd)

Item	Kinetic Energy (Ft-Lb)	Weight (Lb)	Leading Section	Structure/Shield/Barrier
c) Pressurizer				
1) Manway Cover Stud and Nut	71	4-1/4	Solid Circle 1-1/2" Diameter	Pressurizer Enclosure
2) Lower Temperature Element	290	3	Solid Disk 2-3/4" Diameter and 1/2" Thick	Pressurizer Enclosure
d) Control Rod Drive Assembly	57,000	900	Solid Circle 10" Diameter	Missile Shield on Reator Vessel
e) Main Coolant Piping Tempera-ture Nozzle with RTD	1,125	11.1	Solid Disk 2-3/4" Diameter and 1/2" Thick	Secondary Shield Wall
f) Surge and Spray Piping Wells with RTD Assembly	277	1-3/4	Solid Disk 2-3/4" Diameter and 1/2" Thick	Secondary Shield Wall
g) Main Coolant Pump Thermal Well with RTD	1,125	11.1	Solid Disk 2-3/4" Diameter	Secondary Shield Wall and 1/2" Thick
h) Shutdown Cooling Valve Stem	3,340	85	Solid Circle 2-1/4" Diameter	Secondary Shield Wall

TABLE 3.5-1A

CHARACTERISTICS OF TORNADO GENERATED MISSILES

Type of Object	Density (lb/ft^3)	Length (ft)	Minimum Area of Impact (ft^2)	Weight (lb)	Velocity (mph)
4" x 12" Wooden plank	50	12	0.33	248	191
Utility pole 13.5" dia. (1)	43	35	0.99	1505	124
1" Solid steel rod (2)	490	3	0.0054	8.01	132
6" Schedule 40 Pipe (2)	490	15	0.239	285	62
12" Schedule 40 Pipe (3)	490	15	___	___	___

(1) Still under investigation
(2) No higher than 10 ft above the ground
(3) Will not be sustained by the vertical wind

TABLE 3.5-1B

MISSILE DATA

Type of Missile	$C_d A_{max}/W$	Max. velocity (mph)	Elevation (ft)
4" x 12" wooden plank 12' long. Density - 50 lb/ft^3	0.06	229	70
Utility pole 13.5" diameter 35' long. Density - 43 lb/ft^3	0.026	182	26
1" solid steel rod 3' long. Density - 490 lb/ft^3	0.031	192	32
6" schedule 40 pipe 15' long. Density - 490 lb/ft^3	0.029	188	29
12" schedule 40 pipe 15' long. Density - 490 lb/ft^3	0.021	170	21
Automobile	0.026	182	26
2" x 4" x 10' wooden plank. Density - 50 lb/ft^3	0.12	269	130

NOTES:

1) The velocity values presented above represent the maximum theoretical values that could be predicted. They are presented for information only and do not represent a design basis.

2) The values in the last two columns of Table 3.5-1B are obtained using Grand Gulf PSAR Amendment 3, item 3.1, Table 1.

3) The utility pole and automobile missiles originate only at ground level, and all other missiles can originate at any facility elevation. The elevations shown in the last column of Table 3.5-1B are the difference between the elevations at which impact occurs and the elevations at which the missiles originate.

TABLE 3.5-2

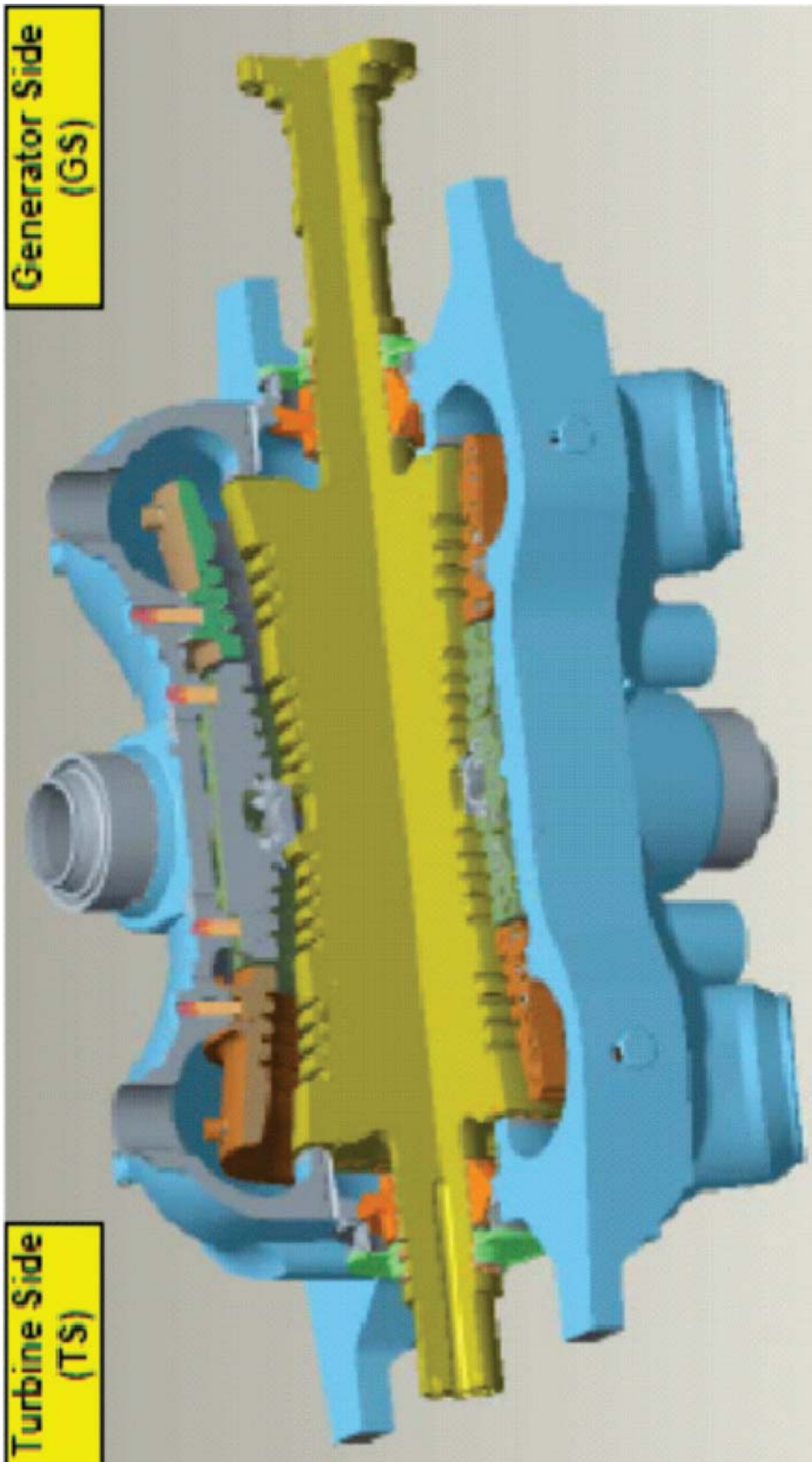
MATERIAL PROPERTIES OF LP ROTOR DISCS

DELETED

TABLE 3.5-3

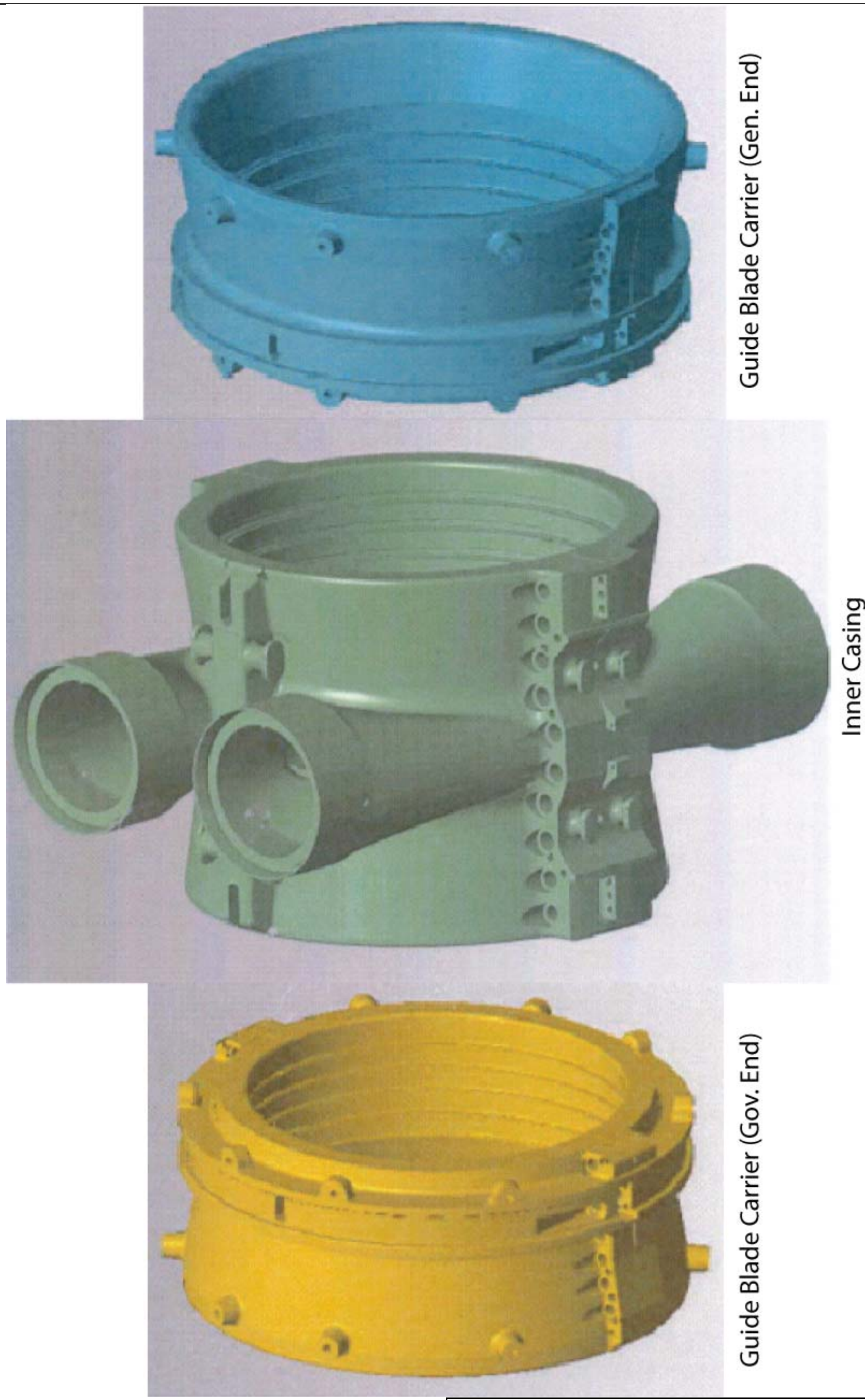
DESIGN OVERSPEED MISSILE EXIT PARAMETERS

DELETED



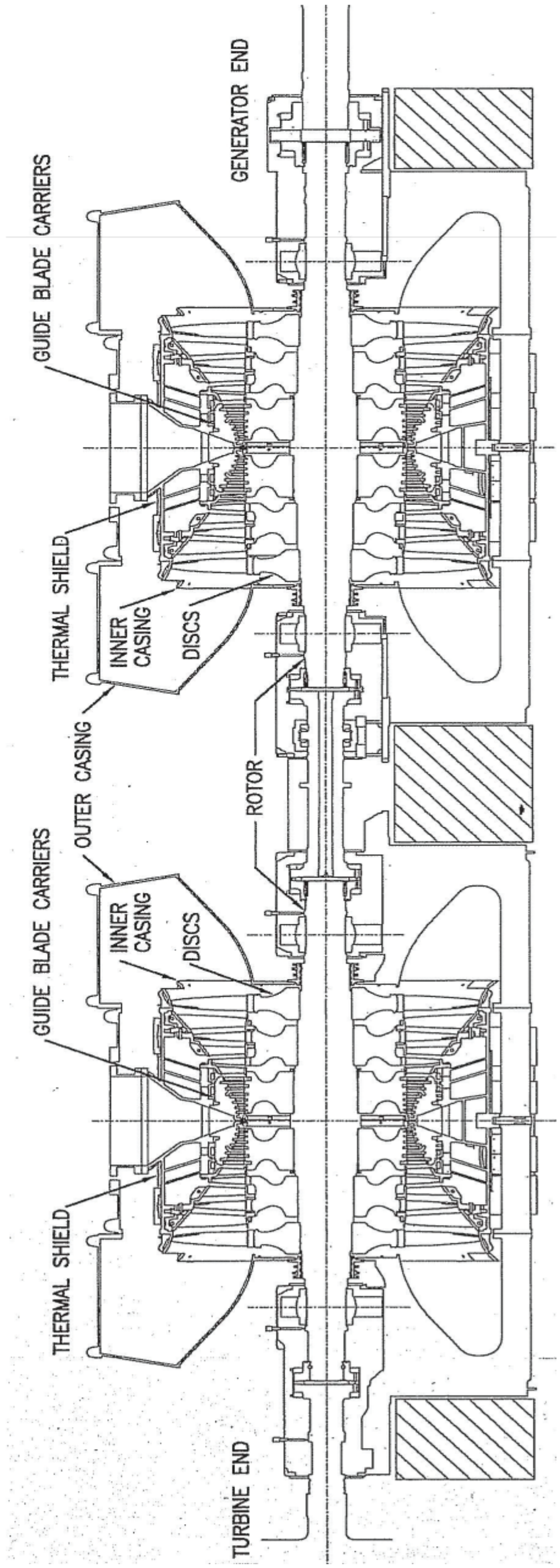
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

HIGH-PRESSURE ELEMENT
1800-RPM DOUBLE-FLOW DESIGN
FIGURE 3.5-1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

INNER CASING AND GUIDE BLADE CARRIERS
FIGURE 3.5-2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

LOW-PRESSURE ELEMENT
1800-RPM DOUBLE-FLOW DESIGN
FIGURE 3.5-3

DELETED

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

TYPICAL LP CYLINDER

FIGURE 3.5-4

DELETED

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

PHYSICAL DIMENSIONS OF TURBINE MISSILE
FIGURE 3.5-5

Refer to drawing
8770-G-548

FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REACTOR BUILDING HATCH COVERS & MISC DET'S – M&R, SH1 **FIGURE 3.5-6**

3.6 PROTECTION AGAINST DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

The criteria for pipe break protection has evolved over time. The initial design bases were in accordance with guidance provided in letters sent by A. Giambusso, AEC Directorate of Licensing, in December 1972. This original guidance was supplemented by the issuance of Regulatory Guide 1.46 in May 1973 for all systems except the Reactor Coolant System. The Reactor Coolant System used different criteria as discussed in Section 3.6.2 The plant was then constructed with pipe whip restraints located and designed to comparable criteria that was less intensive from an analytical standpoint, yet provided more whip restraints than Reg. Guide 1.46 would require. Leak before break criteria was then adopted as an alternative means to treat Reactor Coolant hot leg and cold leg loop piping whip criteria per NUREG-1061. Generic Letter 87-11 was adopted as an alternative means to provide pipe break protection for Class 2, Class 3, and Non ASME Class systems to minimize the addition of or facilitate the removal of excess arbitrary intermediate pipe whip restraints. Regulatory Guide 1.46 was withdrawn in March of 1985 as more current information was provided by the July 1981 revision of the Standard Review Plan, Section 3.6.2. The following sections are comprised of paragraphs pertaining to the evolutions mentioned above.

3.6.1 SYSTEMS IN WHICH DESIGN BASIS PIPING BREAKS OCCUR

Circumferential (guillotine) and longitudinal (slot) breaks were postulated for the RCS hot and cold legs in the original plant design. Since then, however, the NRC revised General Design Criteria (GDC) 4 to include the following statement: “dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping”. The dynamic effects of a LOCA include the effects of missiles, pipe whipping, discharging fluid (i.e., jet impingement), decompression waves within the ruptured pipe and dynamic or nonstatic pressurization in cavities, compartments, and subcompartments. NUREG-1061 established criteria for existing plants to determine which systems were allowed exemption and the methodology that was applicable. Reference 24 demonstrates that the primary loop piping meets all of the criteria for application of leak before break presented in NUREG-1061, Volume 3. As a result, the mechanical/structural loads associated with dynamic effects of guillotine and slot breaks in RCS hot and cold legs are no longer considered a plant design basis (References 26 and 27). The leak before break methodology concludes that:

a. Small cracks which may go undetected during inspections do not grow significantly during service.
b. Cracks which are assumed to grow through the pipe wall would leak significantly while remaining stable. The amount of leakage is detectable with a safety margin of at least a factor of 10.
c. Cracks of the length that leak at the rate in (b) can withstand normal operation and safe shutdown earthquake loads with a safety factor of at least $(2)^{1/2}$.
d. Cracks twice as long as those addressed in (c) will remain stable when subjected to normal operation and safe shutdown earthquake loads.

Although the requirement for designing for dynamic effects associated with a RCS hot or cold leg break have been eliminated from the plant design bases, the original design features installed to mitigate the consequences of such a break have been retained, with the exception of the SG sliding base support, the lower inlet restraint cables and upper RCP whip restraints around the driver mount and portions of the primary shield wall hot leg whip restraints removed for access to replace instrument nozzles (see Section 6.2.1.3.3 for changes to the reactor cavity pressure relief function). As a result of the installation of the replacement steam generators (RSGs) during the steam generator replacement outage, the shim plate attached to the SG sliding base support casting has been permanently removed, thereby deleting the North-South direction LOCA restraint for the SG sliding base support. The following subsections describe the assumptions and methodology used to design the RCS pipe restraints. The environmental qualification design basis for safety related equipment inside containment remains unchanged (Reference 25).

Design basis piping breaks are postulated to occur in the following systems or portions thereof:

a) reactor coolant system (except hot and cold legs)

b) high pressure safety injection system (piping which is part of reactor coolant pressure boundary only)

c) all lines used for shutdown cooling which includes portions of the low pressure safety injection system

d) chemical and volume control system (letdown and charging lines)

e) main steam system

f) main feedwater system

g) steam generator blowdown system

h) auxiliary steam system

The analysis of high energy line breaks outside the containment is presented in Appendix 3C for items e) and f) above. Appendix 3D includes analyses for items c), d), g) and h).

3.6.2 DESIGN BASIS PIPING BREAK CRITERIA

In analyzing the effects of LOCA pipe rupture, both circumferential (guillotine) and longitudinal (slot) breaks are considered capable of occurring at any location along the piping.[(1)] Guillotine breaks and slot breaks with an area up to the cross-sectional pipe flow area are assumed. The effects of resulting pipe whip are considered in the design. The effects for jet impingement resulting from slot breaks are also considered. Piping 1 inch and under is not considered to rupture.

AEC Regulatory Guide 1.46 required that rupture locations for piping systems inside the containment be chosen based on stress limit and usage factor criteria, with a minimum of four such locations analyzed inclusive of terminal ends. The guide further required that protection for pipe rupture (pipe restraints, separation) be provided for piping systems in which operating temperatures exceed 200 F or operating pressures exceed 275 psig. (Note that for any future changes to the plant design, the Standard Review Plan section 3.6 has superseded Reg. Guide 1.46, which has been withdrawn.)

---

(1) See Section 6.2.1.3.3.a.

Pipe whip restraint locations for this plant are chosen based on maximum pipe spans required to develop ultimate moment and torque capabilities of the pipe cross section. Analyses are performed for typical piping configurations using forces developed under slot and guillotine type breaks as shown in Figure 3.6-1. All piping systems with operating pressures above 125 psig are considered in the analysis.

The pipe rupture criteria utilized for restraint of the primary coolant loops is different from other pipe whip criteria. It is as follows:

a) The supports for the steam generators and reactor are designed to accommodate pipe rupture loadings associated with a LOCA (see Section 3.9.1.4.2).

b) The reactor coolant pumps are restrained from becoming a missile in the event of a LOCA.

c) The primary coolant piping in the primary shield wall is restrained to minimize pipe separation in the event of a guillotine rupture of either the hot or cold leg.

The location of the piping restraints in accordance with criterion b) and c) are shown on Figure 3.6-50. The restraints to meet criterion b) include reactor coolant pump suction line stops and around the RC pump motor. The restraints to meet criterion c) include the stops around the primary coolant loops in the primary shield wall and the wall itself. A guillotine rupture of the cold leg at the nozzle of the steam generator is the only leg expected to form a plastic hinge. This is confirmed by the analysis described above. The force will cause the pipe to be driven to the floor.

It should be noted that circumferential (guillotine) and longitudinal (slot) breaks in RCS hot and cold leg piping are no longer considered a design basis for GDC 4 (Reference 27 NRC acceptance letter for leak before break). The primary loop piping is not susceptible to failure from the effects of corrosion, water hammer, fatigue, brittle fracture or indirect causes such as missiles or failure of nearby components. As a result, the mechanical/structural loadings associated with the dynamic effects of a large hot or cold leg break need not be considered.

The criteria described herein used for pipe rupture analyses demonstrate that protection of safety related systems, equivalent to that afforded by the criteria of Regulatory Guide 1.46, is provided since a greater number of restraints are provided on a greater number of piping systems.

For Class 2, Class 3, and Non ASME Class Systems, Generic Letter 87-11 dated June 19, 1987, eliminated the requirement for all dynamic effects (missile generation, pipe whipping, pipe break reaction forces, jet pressurizations and decompression waves within the ruptured pipe) and all environmental effects (pressure, temperature, humidity and flooding) resulting from arbitrary intermediate pipe ruptures. It also allows the elimination of pipe whip restraints and jet impingement shields placed to mitigate the effects of arbitrary intermediate pipe ruptures.

Generic Letter 87-11 revised Branch Technical Position MEB 3-1, "Postulated Rupture Locations in Fluid System Piping Inside and Outside Containment", as contained in the Standard Review Plan (SRP), Section 3.6.2, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," Reference 28. All modifications to Class 2 and 3 piping systems may invoke this new criteria (see Appendix 3.J) in lieu of the original criteria provided that the requirements stipulated in Appendix 3J are fully complied with. The new criteria is based on portions of the revised Branch Technical Position MEB 3-1 applicable to St. Lucie Unit 1.

As stated in NRC Generic Letter 87-011, Reference 27, "Licensees of operating plants desiring to eliminate previously required effects from arbitrary intermediate pipe ruptures may do so without prior NRC approval unless such changes conflict with the license or technical specifications. ...the licensees' updated FSARs should reflect eliminated hardware associated with arbitrary intermediate pipe ruptures." Arbitrary intermediate pipe ruptures, which previously were specified are no longer mentioned or defined. However, requirements for postulated terminal end pipe ruptures, postulated intermediate pipe ruptures at locations of high stress and high usage factor and for leakage cracks are retained.

3.6.3 DESIGN LOADING CONDITIONS

The design loading combinations, design condition categories and design stress limits for safety related piping systems are given in Table 3.9.3.

All safety related equipment is designated in accordance with the rules stipulated in Sections 5.2.1.4 and 3.9.2.

The following loads are considered in the design of the pipe whip restraints inside the reactor containment and reactor auxiliary buildings:

D = Dead load of the pipe whip restraint

P_R = Steam/water jet forces and/or pipe whip reactions resulting from a ruptured pipe. A dynamic load factor of 2.0 is used in the design of all pipe whip restraints to account for the dynamic nature of the load.

DBE=Design basis earthquake load.

The pipe whip restraints are designed to withstand the following load combinations within the allowable stresses specified:

	Load Combination	Allowable Stress
a)	D + DBE	90 percent of material yield strength
b)	D + P_R	90 percent of material yield strength

The configurations and details of typical pipe whip restraints are shown on Figures 3.6-2, 3 and 18.

3.6.3.1 Asymmetric LOCA Loads (historical)

The RCS hot and cold leg piping meets the criteria of 10 CFR 50 GDC 4 and NUREG-1061 for application of leak-before-break methodology (as established by Reference 24). Consequently, dynamic effects caused by circumferential (guillotine) or longitudinal (slot) breaks of this piping need not be considered, and therefore, analysis of the capability of RCS components to withstand asymmetric LOCA loads is no longer required.

As part of the original (and later) plant design bases, the reactor pressure vessel, fuel assemblies and internals, control element assemblies, primary coolant piping and attached ECCS piping, all primary system supports, and the biological and secondary shield walls were analyzed for asymmetric LOCA loads resulting from such breaks (see Appendix 3H). The results of the evaluations are described below.

Pursuant to the NRC request of February 16, 1978, evaluations conducted for asymmetric LOCA loads (see Appendix 3H) were expanded in scope to include an assessment of the reactor pressure vessel, fuel assemblies and internals, control element assemblies, primary coolant piping and attached ECCS piping, all primary system supports, and the biological and secondary shield walls for a spectrum of breaks in the primary system.

The results confirm that the vessel supports will adequately withstand all the loads resulting from the postulated physically possible circumferential break in the vessel inlet pipe. The cold leg guillotine break in the cavity is the break which results in the largest loading of the vessel supports. Therefore the vessel supports are clearly adequate for all other break locations.

Results also show that all supports for the primary system are adequate for all break locations, that the stresses in the intact primary piping and attached lines are sufficiently low to ensure performance of intended functions, and that the biological shield wall performs its intended function. The secondary shield wall is designed for postulated primary system ruptures within the steam generator subcompartments.

Results further show that the control element assemblies, which are not needed for the postulated breaks, maintain the pressure boundary integrity when subjected to the motions resulting from those breaks. Analyses of the adequacy of the reactor internals and fuel were performed using a slightly different hydraulic model to compute the forcing functions to be applied to the said internals and fuel. The forcing functions took no consideration of limitations in break area other than may be due to the strength of the primary piping itself. This was done to take advantage of these very same analyses performed for a virtually identical plant, Calvert Cliffs, recognizing that results thus obtained would bound those that would be computed for the breaks physically possible in St. Lucie 1. The results of the reactor internals analyses demonstrate that the internals are capable of accommodating loads resulting from the largest cold and hot leg breaks. The analyses of the fuel indicates that for an unrestrained inlet break localized crushing of the grid spacers occurs near the top of some of the peripheral assemblies. However, this limited crushing does not prevent maintaining the coolable geometry of the assembly and the core overall. See Reference 22.

In Reference 23, the NRC staff concluded that there is reasonable evidence that the Unit 1 reactor would withstand the effects of asymmetric LOCA loads.

3.6.4 DYNAMIC ANALYSIS

3.6.4.1 Pipe Whip

The reaction force on the ruptured pipe and the force resulting from jet impingement are calculated as follows:

$$\text{Reaction Force} = kP_o A \text{ (lb)}$$

$$\text{Jet Impingement Force} = kU_m P_o A \qquad (1)$$

where k is the thrust coefficient which is used to compute the impingement force, and U_m is the ratio of the peak axial velocity at some distance "x" from the origin of the jet to the initial jet velocity. The peak velocity at any distance from the origin of the jet is obtained from equations and charts of Reference 1.

P_oA is the product of the initial pressure inside the broken piping and the break flow area.

The thrust coefficients used for guillotine and slot breaks are as follows:

THRUST COEFFICIENT, k

Type of Break	Dry Saturated Steam	Two Phase, Flashing Water or Wet Steam	Sub-Cooled Water
Guillotine	1.01	1.12	1.6
Slot	0.63	0.68	0.97

The design for the protection against dynamic effects of pipe rupture is based on the premise that breaks could occur anywhere in the pipe run, and not on the criteria formulated on the hypothesis that satisfactory protection from whipping pipes can be achieved by restraining against breaks postulated at stress related locations. Consequently many more restraints are utilized than would have been provided had the stress related criteria been used. The spacing of such restraints is based on the ultimate moment carrying capability of the restrained pipe such that for a break between consecutive restraints additional breaks would not result at either restraint or any other one. As a result, the protection provided against pipe rupture is at least the equivalent of that which would be achieved according to the stress related pipe break criteria and the pipe whip analysis methodology of Reference 1. Even assuming that an individual restraint could not bear the entire pipe rupture load, the adjacent restraints would share the load and still result in an adequately restrained system.

To verify whether the restraints are adequately designed, since the design is based on a static force applied with a dynamic load factor of 2.0, it is necessary to examine the dynamic characteristics of the force as well as the structural response of the restraint.

The dynamic behavior of the reaction force on any given pipe and any given break depends on several factors among which the more salient are:

a) characteristics of contained fluid (steam, compressed water, pressure, temperature)

b) capacity of reservoir behind break

c) configuration of piping system

d) location of the break with respect to the reservoir (friction effects)

e) presence of flow restrictors in the line.

The initial peak value of the reaction force is only slightly affected by friction effects, phase change effects and flow restrictors. However, the subsequent transient phase is very much affected by the above factors and also the actual configuration of the system. Thus, for each given break, it is far easier to predict fairly accurately the first peak value of the reaction force than it is to predict the subsequent transient phase. From experiments (Reference 2) and also published literature (Reference 3), the following can be stated with regard to this transient regime:

a) For subcooled decompression (such as would occur in feedwater line, shutdown lines, rain coolant lines), the initial force peak decays rapidly as the pressure drops to the saturation pressure for the given temperature, then stabilizes at a slowly decaying value, as shown in Figures 3.6-5 and 3.6-6. The duration of this sharp transient is dependent on the location of the break with respect to the reservoir, the presence of any flow obstacles between the reservoir and the break, and the friction in the line. In general, the duration will be less than 1-2 milliseconds.

b) For steam breaks, again the peak force value lasts only a few milliseconds. Prior or subsequent values of the force are lower, although the effect is not as pronounced as in subcooled decompression.

Since, due to the gap between piping and restraint, the broken pipes also require times of tho order of milliseconds to strike the restraints, the peak force will not be acting on the break constantly, but rather the total impulse received will be less than that which would he calculated by using the peak force as constant. Furthermore, the energy imparted will be even less, e.g., half the impulse corresponds to one quarter of the energy. Thus a constant force of lower magnitude could be justified in a static analysis.

The dynamic load factor (DLF) is the product of the force actually acting on the restraint times a dynamic amplification factor accounting for the response of the restraint, the maximum value of which would be 2.0 if the restraint behaved entirely elastically. In reality the restraint will respond elasto-plastically and the magnitude of the dynamic amplification factor (DAF) will be less than 2.0. The force acting on the restraint is the sum of the jet force acting at the time of impact (transmitted through a lever arm) plus the force acquired by the pipe due to its acceleration through any existing gap between the pipe and the restraint.

Hence: $F_{\text{equiv static}} \times DLF = (F_{\text{jet at impact}} + F_{\text{acceleration through gap}}) \times DAF$ (3)

The peak thrust values used in the static analysis are essentially the peak theoretical values assumed in the methodology of Reference 1, corrected to account for friction effects which always exist as flows develop and continue. In case of slot breaks the exit friction losses become very significant, thus even if the initial thrust (no flow) equals $P_O A$, its duration is such that the impulse may be negligible with respect to the total impulse developed under flow conditions.

In the final analysis, quantitative justification of the coefficient used to predict jet thrust and the correctness of the DLF employed can only be provided by performing a dynamic analysis of the piping-restraint system. To determine the degree of conservatism provided by the static analysis method employed, a number of systems will be dynamically analyzed and the results will be compared to those obtained by the static method. Refer to Section 3.6.4.3.

A diagram for each of the piping systems inside and outside of containment that is postulated to rupture and for which restraint is necessary is presented in Figures 3.6-10, 3.6-13 through 3.6-18, Figure 3.6-29, and Figures 3.6-33 through 3.6-51. As shown on these figures, the constrained direction is perpendicular to the pipe axis.

3.6.4.2 Jet Impingement

Curves have been developed for lines which run in proximity of safety related components. These curves are used to determine jet impingement forces resulting from pipe breaks and acting on adjacent equipment, piping and structures. For flat surfaces, the jet impingement forces on the analyzed components are calculated by multiplying the jet impingement pressure by the area of that component which is contained within the are a of the jet.

The geometry of the jet, its pressure distribution and its temperature distribution depend on the nature of the discharged fluid and the sur-rounding medium. For a single phase jet (dry steam) the divergent angle of the jet is calculated as a function of the degree of heating (ratio of the jet temperature in °K to temperature of surrounding air in °K) from charts developed in Reference 4 and reproduced here in Figures 3.6-7 and 3.6-8. For a two phase jet, the divergent angle is a function of densities of the jet mixture and the surrounding medium and can be calculated from Figure 3.6-9. For a subcooled water jet, the expansion angle is taken as 25°.

The jet force on a given surface depends on the extent and shape of the surface. As long as the struck surface is larger than the jet, equation (1) is used. This equation is conservative in that no credit is taken for the velocity profile of the jet at every point.

Abramovitch has shown that the velocity profile at any point "x" along the jet axis is given by:

$$\frac{U(x)}{U_m(x)} = \left[1 - \left[\frac{y(x)}{d(x)} \right] \right]^{1.5} \qquad (2)$$

where d(x) is the jet diameter at point "x".and y(k) is the distance from the axis to the jet at point "x", at which U(x) is measured.

Equation (1) calculates the jet force on a structure as if the average velocity at "x" corresponded to the peak velocity at "x", whereas, in reality the average velocity is only 0.26 $U_m(x)$, and the corresponding average force is 0.52. The average force on a wall intercepting the entire jet is thus overestimated by using equation (1).

If the struck surface does not intercept the entire jet, the impingement force is further reduced to account for the projected area of the target. If the struck surface is very small with respect to the jet and such that the jet completely surrounds it, the force is equal to the drag force on the target and is calculated by multiplying equation (1) by the proper drag coefficient.

For pipes, the jet impingement force on the analyzed components are calculated by multiplying the jet impingement pressure by the projected area of that portion of pipe which is contained in the area of the jet and multiplying the result by a factor of 0.6 to account for the curvature of the component.

In evaluating the load carrying capability of pipe the internal pressure and the effects of strain hardening are considered. In the case of a simply supported pipe experiencing a slot failure, the cross-sectional area of the pipe capable of sustaining a moment is considered to be the defect area which accounts for the presence of the slot break instead of the original pipe area.

3.6.4.3 Pipe Whip Analysis - Main Steam and Feedwater

As shown on Figure 3.6-52, a break location was established at node 12 for the main steam line. A break at node 12 results in the maximum impact at the restraint located at node 9 and the maximum total strain in the pipe.

Span lengths between pipe whip restraints are as shown on Figures 3.6-52 through 3.6-55,for main steam and feedwater piping. The maximum span lengths depicted were established using the design criteria presented in Section 3.6.5-1. As stated in 3.6.5.1, failure stress is limited to that value which corresponds to 50 percent of the true ultimate strain when related to a simplified stress-strain curve (Figure 3.6-9A).

For the steam line break selected (node 12 on Figure 3.6-52), the moment required for full plasticity, for yielding and the actual moment computed for that limiting case are 35.2×10^3 in-kip, 27.8×10^3 in-kip and 35.2×10^3 in-kip, respectively. The actual computed moment and the moment to full plasticity are equal since for this limiting span length the pipe does plastically deform, but does not whip.

Sensitivity studies for the main feedwater line outside containment are summarized for variations in gap length and pipe wall thickness in Table 3.6-1 and Figures 3.6-63 through 3.6-68. A reduction in gap reduces peak restraint reactions while decreasing wall thickness seems to increase reactions.

The span method of restraint placement (See Section 3.6.5.1) does not provide for a margin to full plasticity since the method itself assumes the pipe to go plastic. The span method does, however, prevent maximum calculated strain from exceeding one-half of the ultimate strain. For example, at the instance when a pipe is fully plastic, the pipe can still carry moments - only additional strain energy into the pipe will cause further straining up to the ultimate.

If the strain hardening is ignored, as was done in this analysis, then the span method does not predict the strain corresponding to imposed moment since this strain is not unique. However, Figure 3.6-61 shows that the maximum strain does not even approach half ultimate strain values. Since zero strain hardening was employed, it follows that the calculated moment at maximum strain and the moment required for full plasticity are identical, 35.2 in-kip.

At this moment value, the pipe has not collapsed and can continually carry equal or diminished loads. The moment necessary to yield the outer fibers of the pipe is 27.8 in-kip.

Figures 3.6-52 through 55 provided the span lengths, restraint locations and node locations using nodal breakdown requirements for pipe whip analysis. Nodal breakdown requirements for pipewhip analysis are different than those required for stress analysis as reflected in the piping isometrics of Section 3.6.

Because node points were included in the numbering scheme in Figures 3.6-52 through 55 and since nodes were not shown in other Section 3.6 figures, no correspondence should be expected between the two except that piping dimensions and location of pipe whip restraints are identical.

To illustrate this point refer to Figures 3.6-36 and 3.6-52. Figure 3.6-36 restraint locations MS-2, MS-3 and MS-5 correspond to restraint locations 2, 4 and 8 in Figure 3.6-52, respectively. All the related figures have been reviewed for accuracy.

a) <u>Conclusions</u>

Four cases (2 feedwater line breaks and 2 main steam line breaks) of circumferential pipe rupture were analyzed for maximum restraint reactions and maximum pipe strain. All analyses were extended for a period of 0.2 seconds past initiation of pipe rupture when steady state oscillations of the deflection of the rupture point occurred with decreasing amplitude. Strain hardening in the pipe was assumed to be zero for conservatism.

Figure 3.6-56 indicates that blowdown forces for the feedwater line rupture reach a steady state value of 110,000 lbs at 0.034 seconds; for the main steam line, the blowdown forces reach a value of 135,000 lbs at 0.1 seconds, decrease exponentially to 100,000 lbs at 0.15 seconds, and continue to decrease at the same exponential rate thereafter.

Since the steam line is multi-planar, restraint reactions can occur in more than one direction and in more than one restraint. This is evident from the reaction force results plotted on Figures 3.6-57 and 3.6-58. Peak reactions in all cases except two were below the 2 KPA factors applicable to the line under analysis. In the two exceptions (feedwater line break at node 6, Figure 3.6-59 and main steam line break at node 12, Figure 3.6-58), the peak duration is approximately 0.002 seconds or less.

Since the natural periods of the restraint system, consisting of the steel frame restraints, the embedments and the concrete wall are of 0.002 seconds or less, this system was reviewed to determine to what extent, if any, its Primary function of pipe restraint during blowdown may be impaired.

A very conservative analysis of the steel frames, in which the stiffening effects of collar plates and webs were ignores and the pipe whip impulse loads were treated as step functions constant in time, revealed that yield would occur in the structure. Since the pipe whip dynamic analysis was based on the assumption of elastic, non yielding restraints, the effect of yielding would be to reduce the loading peaks shown in Figures 3.6-58 and 3.6-59. However, the nonyielding assumption used in the pipe whip analysis is taken, as the more conservative approach. In either case, yielding or nonyielding, the steel frames will perform their function of adequately restraining the pipes against excessive movement.

Assuming complete rigidity of the steel frame restraints and concrete, the bolts, subjected to a pulse (hat function) loading of 0.002 seconds duration were shown to reach a peak strain of 0.0155 in/in (based on a bilinear stress-strain curve in which Young's modulus $E=30x10^6$ psi and the strain hardening modulus S-0.05E) for carbon steel. This strain is well below 1/2 ε_μ for carbon steel (taken as 0.1) and is confined to the threaded portion of the bolt. Once more it is seen that under very conservative assumptions the bolts do not rupture, and that yielding results in lowering the applied pulse peaks due to pipe whip.

Assuming, once more, that the restraint frames remain rigid, and that a step function load is applied through the bolt anchor plates to the concrete with a peak equal to the applied pipe whip impact-pulse distributed to the embedded bolts, it was shown that the concrete would not fail in shear (i.e., pullout) and that the concrete wall was adequate to resist these loads.

In summary the design of the pipe whip restraints and embedments is considered adequate to perform their primary function of limiting pipe motion and secondary damage following a pipe break because there will be no intolerable loadings as a result of exceeding the factor 2.0 for the k load factor.

Maximum strains in the feedwater and main steam lines are found by adding the yield strain to the maximum plastic strain. These peak strains, in all cases, are considerably less than half the ultimate strain of the materials (main steam - steel, A 155 GR-KC 65; feedwater - steel, A 106 GRB).

Critical results of these analyses are indicated on Figures 3. 6-57 through 3.6-64 and on Table 3.6-2 for the four ruptures considered.

b) Dynamic Analysis

Four different breaks were analyzed, two on a main steam line and two on a feedwater line. The break locations chosen, restraint locations, geometry, material properties, and maximum operating temperatures and pressures for each break condition are shown in Figures 3.6-52 through 3.6-55. The four breaks were chosen as representative of breaks producing maximum impact reactions and pipe strains. In all cases circumferential breaks were analyzed since the greatest potential for whipping a pipe exists. Thrust forces, at the break locations were developed by performing a time history, thermal hydraulic analysis of the blowdown with the RELAP-3 code (Reference 14), suitably modified to predict thrust forces.

With the RELAP-3 code the transient energy, momentum, and state equations were solved for an assembly of control volumes and flow paths modeling the piping system. The total thrust out of the break was evaluated as the sum of three components:

a) A Momentum flux component equal to $W^2/g_c \rho A$, representing the outflow of moment out of the control volume about the break,

b) a pressure force component equal to $(P_e - P_a)A$, representing unbalanced pressure forces on the control volume about the break, such as occurring when flow is choked at the exit plane, and

c) an inertial component equal to $\left[\frac{W_{t+\Delta t} - W_t}{g_c \Delta t}\right] L$

representing the thrust due to acceleration caused by the change of momentum with time within the control volume about the break.

Herein t is the time, W the mass flow rate, A the break flow area, P_e the pressure at the exit plane, P_a the ambient pressure, ρ the fluid density, g_c the gravity constant, and L the length of the control volume chosen to represent the break. The velocity used in the calculations is either the inertial velocity (Bernoulli's equation) or the choking velocity as found from Moody's critical flow correlation (Reference 12).

Time dependent blowdown forces are shown by curves in Figure 3.6-56, Empirical functions conservatively approximating this data, shown in heavy lines on Figure 3.6-56 were used in the dynamic analysis pipe whip program "PLAST" as input. Gap data and spring constants for pipe whip restraints are as indicated in Tables 3.6-3 through 3.6-6 for the four breaks chosen.

The "PLAST" program models the pipe run as a lumped parameter system with elastoplastic material properties. The equations of motion of the system are solved by a step by step integration method in the time domain using varying time steps to insure solution stability.

A pipe run is modeled as a lumped parameter system consisting of discretized "masses" and "springs". The "masses" are represented by the physical mass and rotary inertia of the pipe while the "springs" are represented by pipe stiffnesses corresponding to the 6 degrees of freedom for every point along the pipe axis.

A section of pipe bounded by lumped masses at each end is defined as an "element". A 12 x 12, symmetric stiffness matrix may be written for each such element. The individual terms of the matrix may be represented by the symbol "k_{ij}" where the i,j subscripts refer to the row and column locations respectively of the term within the matrix. For a linear pipe element, the non-zero terms are given below:

$k_{11} = AE/L$

$k_{17} = -k_{11} = k_{71}$

$k_{22} = 12\ E\ I/(L^3 = L\ C)$

$k_{26} = k_{62} = -\ 6\ E\ I/(L^2 + C)$

$k_{28} = k_{82} = -\ k_{22}$

$k_{2,12} = k_{12,2} = k_{26}$

$k_{33} = k_{22}$

$k_{35} = k_{53} = -k_{26}$

$k_{39} = k_{93} = -\ k_{33}$

$k_{3,11} = k_{11,3} = k_{35}$

$k_{44} = GI_x/L$

$k_{4,10} = k_{10,4} = -k_{44}$

$k_{5,5} = 4\ E\ I/L - \alpha$

$k_{5,9} = k_{26} = k_{9,5}$

$k_{5,11} = k_{11,5} = \beta$

$k_{66} = k_{55}$

$k_{68} = k_{86} = -k_{26}$

$k_{6,12} = k_{12,6} = \beta$

$k_{77} = k_{11}$

$k_{88} = k_{22}$

$k_{8,12} = k_{12,8} = -k_{26}$

$k_{99} = k_{33}$

$k_{9,11} = k_{26}$

$k_{10,10} = k_{44}$

$k_{11,11} = k_{55}$

$k_{12,12} = k_{66}$

<u>When</u> E = Young's modulus

G = Shear modulus

L = Element length

A = Cross - sectional area of pipe metal

I = Cross - sectional moment of inertia in bending

Ix = Torsional moment of inertia

$C = 24\ \mu\ (1 + \nu)r^2$

ν = Poisson's ratio

μ = Shear factor (2 for pipe)

$r = (I/A)^{½}$

$\alpha = (3C/L^2 + c)(EI/L)$

$\beta = 6EI/ (L + c/L) – (4EI/L)(4 L^2 + c/4(L^2 + c))$

This stiffness matrix includes the effect of transverse as well as Torsional shear.

Stiffness matrices have also been developed for curved and "stepped" elements with appropriate "flexibility" factors applied.

The equations of motion for an element are written:

$$[M]\{\ddot{X}\} + [C]\{\dot{X}\} + [K]\{X\} = \{F\} \qquad (1)$$

for linear elastic behavior where:

[M], [C], [K] are the mass, damping and Stiffness matrices.

$\{X\},\{\dot{X}\},\{\ddot{X}\}$ are the displacement, velocity and acceleration vectors and {F} is a vector of forces acting at the element nodes (end masses) that keep the element in equilibrium. Hence in the absence of external applied forces at a node these represent internal forces.

The equations of motion for the overall structural system are obtained after adding individual element stiffness matrices (referred to overall global axes). The overall equations of motion have the same appearance as (1) but {F} represents a vector of forces acting on but external to the structural system.

Damping:

An upper bound damping factor[16] is found for the range of periods between 0.025 seconds and the largest system period. This requires the determination of two constants α and β such that:

$$C_{ij} = 2\,\beta M_{ij} + \alpha k_{ij} \qquad (2)$$

where: C_{ij}, M_{ij}, k_{ij} are the damping, mass and stiffness terms of the

$i^{\underline{th}}$ row and $j^{\underline{th}}$ column of the [C], [M] and [K] matrices respectively.

The damping matrix represented in (2) represents a conservative estimate of the system damping.

Plasticity:

The materials that make up a piping system are considered to yield according to Von Mises[17]criteria and are either elastic - perfectly plastic (zero-hardening) or harden "isotropically."[18] The constitutive laws are considered to be "incremental" in that they relate increments of plastic deformation to total stress at a point in body as follows:

$$\Delta\varepsilon'_{ij} = \frac{\Delta\ \tau'_{ij}}{2\,G} + \frac{\partial F(\ \tau'_{ij})}{\partial\tau_{ij}} \qquad (3)$$

where: Δ ( ) = increment of ( )

G = Shear modulus

ε'_{ij}, τ'_{ij} are the tensor components of strain and stress deviators respectively

$F(\tau'_{ij})$ is defined by:

$$F(\tau'_{ij}) - J_2 = 0 \tag{4}$$

The Von Mises yield criteria are stated in the following way:

$$\text{I F} \quad \left\{ \begin{array}{l} F > J_2 \\ \Delta F > o \end{array} \right\} \text{ then yielding occurs}$$

$$\text{I F} \quad \left\{ \begin{array}{l} F \leq J_2 \\ \Delta F < o \end{array} \right\} \text{ then elastic action occurs}$$

Equation (4) may be depicted as a right circular cylinder making equal angles with three principal axes representing principal stresses at a point. [18] When yielding occurs, the plastic strain increment can be plotted on the same set of axes (with an appropriate scale factor) as a vector normal to the cylindrical surface. Isotropic hardening is represented as an expansion of the cylinder cross - section about its origin (i.e., isotropically). In order to establish hardening parameters it is necessary to know the slope and shape of the uniaxial stress - strain curve beyond initial yielding.

The flow rule associated with the Von Mises yield criteria is not applied directly to the framed structures. Instead, use is made of yield surfaces in "force space" [19] derived from the Von Mises yield surface. These surfaces are symmetrical with respect to principal force axes but lack point symmetry. A simplifying approximation to such surfaces may be made by utilizing their circumscribing sphere in force space. The equation of this space is given as:

$$\Phi = \left(\frac{My}{My^P}\right)^2 + \left(\frac{Mz}{Mz^P}\right)^2 + \left(\frac{Fx}{Fx^P}\right)^2 - k^2\left(1 - \left(\frac{Mx}{Mx^P}\right)^2\right) \tag{5}$$

<u>Where:</u> Mx, My, Mz are the internal moments about the designated axes.

F_x^P, M_x^P, M_y^P, M_z^P are the fully plastic values of these forces

Fx is the internal axial force on a member.

κ is the sphere radius.

The associated flow is then:

$$\Delta X_i^P = \lambda \frac{\partial \Phi}{\partial Fi} \tag{6}$$

Where: $\Delta X_i^{\,P}$ = the $i^{\underline{th}}$ component of plastic displacement increment

F_i = the $i^{\underline{th}}$ component of internal force (limited to the components in eq. (5))

The total displacement of a point can be broken up into elastic and plastic components:

$$\Delta X_i = \Delta X_i^{(e)} + \Delta X_i^{(P)}$$

Then since only elastic displacement contribute to force at a node, equation (1) becomes: (7)

$$[M]\{\ddot{X}\} + [C]\{\dot{X}\} + [K]\{X\} = \{F\} + \{F^{(P)}\} \tag{8}$$

Where: $\{F^{(P)}\} = [K]\{X^{(P)}\}$

is the "plastic correction force" developed internally for a yielding element, and assembled as a total correction force for the overall structure system. Expressions for Λ in equation (9) for cases of one and two nodes of an element yielding are shown in detail in reference (20). These were based on the following assumptions:

a) - Small deformation
b) - Concentrated forces applied only at nodes (masses)
c) - Yielding at a cross - section occurs simultaneously over the entire cross - section or not at all
d) - There is no spread of yielding beyond the node along the beam axis
e) - The flow rule of eq (6) applies

In the case of isotropic hardening, one seeks parameters that indicate the correct yield surface to use (in force space) for the flow rule, eq. (9). Towards this end the following terms are defined for pipe beams:

a) - Effective stress:

$$\sigma_e \equiv \sqrt{3J_2}$$

where:

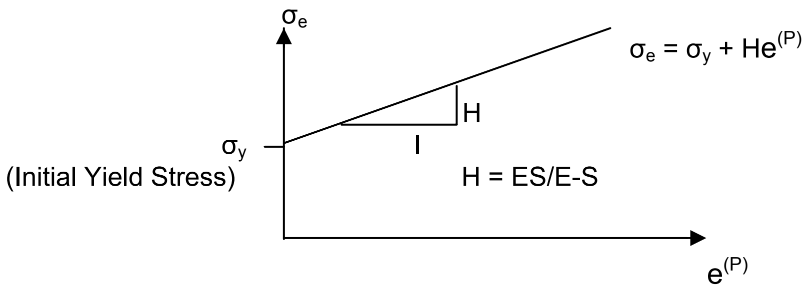
$$J_2 = 1/3\,(\tau_{xx})^2 + (\tau_{xo})^2$$

b) - Effective plastic strain:

$$e^{(P)} \equiv \sum_{i=1}^{n} (\Delta e_i^{(P)})$$

Where: $$\Delta e_i^{(P)} \equiv \sqrt{2/3\,[(\Delta\varepsilon_{xx}^{(P)})^2 + 2(\frac{\Delta^y xo}{2})^2]}$$

c) - Effective stress strain curve



E=Young's modulus
S=Plastic modulus

(Shown for a bilinear material)

Fig 1

The effective stress-strain curve is a plot of stress vs plastic strain for a uniaxial specimen. For a three dimensional analysis, it represents the radius of the yield surface plotted against effective plastic strain. The area under this curve is the plastic work of deformation. Therefore if the forces that give rise to yielding along this curve are known the expression for the yield surface in force space may be used to find plastic displacement increments as in equation (6).

Solution

Solution of equations (8) is by the Newmark "Beta" Method [21] using Beta = 1/6 and a convergence rate of 0.1. The initial integration step is found internally as a fraction of the approximate value of the lowest period of the system. Solution stability is assured by maintaining an upper bound of 1/5 on the value of this fraction. Further improvements in the time step may be made by accounting for the lowering of natural frequencies of the system that result from yielding.

In propagating the solution through the time domain, no modifications are made to the initial stiffness and mass matrices. The problem, in short, is considered to be in the "small deformation" regime. However small deformations give rise to large deflections and rotations. Hence, blowdown forces at severed pipes are made to follow the pipe movements. Gaps at restraints, are treated as step changes in displacement force boundary conditions; i.e. a node initially with a zero force specification in some direction suddenly changes its specification to zero displacement in that direction. Pipe whip restraints are modeled as bilinear, elastoplastic springs of zero length and negligible mass. These "take a ride" with the whipping pipe until the gap is closed.

As each element of the system in loaded, it deforms according to elastic and than elastoplastic constitutive laws. Unloading occurs elastically leaving a residual "plastic displacement" in each of the yielded elements.

Restraint models consist of one or more "anchors" and elastoplastic external springs with initial gaps; the latter representing the pipe whip restraints. All hangers and earthquake restraints are considered to have failed. Rebound velocities and impact forces are affected by gap size, restraint and pipe material properties and system damping as follows.

Rebound velocities and impact forces at a restraint result from the instantaneous introduction of a displacement boundary condition at the restraint. This boundary condition imposes displacement constraints on a mass which has the effect of applying external forces on the mass. If the boundary condition nullifies displacements in any direction, the restraint is considered "rigid." If displacements are a linear function of themselves, the restraint is considered "elastic." If displacements are governed by laws of one dimensional elastic-plasticity, the restraint is considered as an "elasto-plastic, strain-hardening" restraint. The sum of the rate of change in momentum of the mass due to the introduction of this boundary condition plus the viscous forces in the mass plus the internal force of the attached pipe is the total force acting on the mass. The resultant total momentum change accounts for the instantaneous rebound velocity of the mass. These factors depend on the mass velocity at impact, the type of restraint (rigid, elastic or elasto-plastic), system damping and the stiffness properties of the pipe. Gap size affects the mass velocity at impact.

The analysis, in each break case, was allowed to run until it was observed that the loaded mass point oscillated with decreasing amplitude about some displacement value. It was noted, in all cases, that the first peak in reaction force magnitude was never subsequently superceded, even though displacement peaks were reached after the reaction peaks. Strain hardening in the piping system was taken as zero for conservatism.

The peak plastic strains in each system are indicated in Figures 3.6-62, 64, 66 and 68. The total strain in each case can be obtained by adding the yield strain to the maximum plastic strain and is seen never to approach $\varepsilon_u/2$.

Results of pipe whip analyses performed on guillotine breaks in two locations of the feedwater line on either side of the penetration and in two locations of the main steam line inside containment are shown in Table-3.6-2. In addition, sensitivity studies for the feedwater line outside containment are shown for a change in gap and a change in wall thickness. A reduction in gap reduces peak restraint reactions as was expected. Decreasing the wall thickness however seems to have the opposite effect. The results of this sensitivity study are given in Figures 3.6-63 through 3.6-68 and Table 3.6-1.

The coefficients used to calculate the jet thrust force on the ruptured pipe are based on the following:

1) Maximum theoretical values of the thrust coefficients have been predicted by Moody (Reference 12) under steady flow conditions to be 1.26 for steam and flashing water, and 2.0 for subcooled water. These values ignore frictional effects in the pipes and exit effects.

2) In real fluids friction and exit losses are present, therefore the maximum theoretical coefficients have been modified to account for such losses. Thrust forces for several piping breaks involving steam and feedwater were derived by using the RELAP-3 thermal hydraulic code (Reference 14). The results show that at steady state flow conditions peak values of the thrust coefficients are close to unity for steam and 1.1 for flashing water. The particular values listed above of 1.01 and 1.12 respectively for steam and flashing water were derived for conditions typical in power plants using 350 psi exit pressure for saturated steam and low quality (≈ 1 percent) and 800 psi pressure at the exit plane for feedwater.

3) For subcooled water frictional effects are accounted for by utilizing a resistance coefficient of 1.23 (Reference 11)

4) To account for the more severe contraction present in the case of a slot break, a contraction coefficient of 0.61 has been chosen (Reference 13).

c) Static Analysis

The actual force acting on the restraint at the time of impact is equal to the sum of the jet force acting on the broken pipe at that time (transmitted through a suitable lever arm) and the force due to the energy acquired by the pipe as a result of its acceleration through any existing gap between the pipe and the restraint.

If this force is to be applied statically, then it must be multiplied by a suitable dynamic amplification factor (DAF) which accounts for the response of the restraint and the structure supporting the restraint. The maximum value of the DAF for a one degree of freedom system is 2.0, assuming the restraint behaves entirely elastically. In reality the restraint will respond elastoplastically, and the magnitude of the DAF will be less than 2.0. Thus, the equivalent static force applicable to any restraint is given by:

$$F\begin{pmatrix} equiv \\ static \end{pmatrix} \le \left[F\begin{pmatrix} jet\ at \\ impact \end{pmatrix} + F\begin{pmatrix} acceleration \\ through\ gap \end{pmatrix} \right] x\,2.0 \tag{3}$$

Since neither the F (jet at impact), nor the F (acceleration through gap) are known (both being functions of piping configuration, gap size, pipe size, break location, etc.) it is convenient to express the equivalent static force by the product of the peak thrust force ($F_{\substack{jet \\ peak}}$ kpa) and a load amplification factor (DLF) which accounts for load increase due to acceleration through the gap and dynamic response of the system. Hence:

$$F\begin{pmatrix} equiv \\ static \end{pmatrix} = F\begin{pmatrix} jet \\ peak \end{pmatrix} \ x\ DLF \tag{3a}$$

therefore, the relation between DLF and DAF is given by:

$$DLF = \frac{DAF}{F\begin{pmatrix} jet \\ peak \end{pmatrix}} \left[F\begin{pmatrix} jet\ at \\ impact \end{pmatrix} + F\begin{pmatrix} acceleration \\ through\ gap \end{pmatrix} \right] \tag{3b}$$

Since DAF was expected often to be equal to unity, a value of 2.0 was chosen for DLF.

An expansion angle of 25^0 was chosen for subcooled water jets as a result of work done in Reference 15. Values of the dispersion angle of subcooled water jets are reported in the literature to vary between 6.5 to 12.5 degrees (half-angle). [(8)] For equal jet exit conditions, the assumption of a smaller dispersion angle would result in higher target loading at equal distances from the jet origin. This is also true for close up targets where total loads would be nearly the same but the distribution of pressure loading would be more severe.

The significant difference between loadings calculated by using a 10 degree or a 12.5 degree half-angle is noted for remote targets only if the actual velocity distributions are used at the target location. However when uniform velocity distribution is utilized having the centerline velocity value, then the choice of a wider dispersion angle becomes conservative:

For instance, for an axisymmetric jet, the centerline velocity at any distance x from the jet origin is given for the conditions of uniform fields of density and velocities at the initial cross section of the jet, and the pole of the main jet region resting at the initial cross section of the jet by:

$$\frac{U_m}{U_o} = 12.4\,\frac{r_o}{x}$$

where:

r_o = radius of the jet at the origin

U_o = exit velocity

The centerline velocity equation contains its dependence on dispersion angle through the factor 12.4, which is given as 2.73/tan 12.5 degrees. For the 10 degrees half-angle, the corresponding velocity expression would be:

$$\frac{U_m}{U_o} = \frac{15.3\,r_o}{x}$$

The velocity, and hence the target loading calculated with a normal velocity distribution, determined for the smaller angle would be approximately 20 percent higher than the corresponding quantity evaluated for the 25 degrees angle. However, the calculated loading based on a uniform peak velocity is conservative by approximately 50 percent.

The-friction factor used in the RELAP-3 program, used to generate blowdown data, is the following:

a) For Junctions having no initial flow

$$k = \frac{f\;A_w}{2(144\,g_c)\,A^3}$$

where:

f	= Fanning friction factor
A_w	= Wetted wall area
A	= Cross sectional area of flow

b) For junctions having flow, the code calculates the friction factor internally from

$$k_j = \left(\frac{P_i - P_{i+1} + \Delta P_{pump,j}}{W_j^2} \right) P_j$$

where:

W_j and P_j are the mass flow rate and density of the fluid at junction j, and $\Delta P_{pump,j}$ is the pressure head due to a pump located at junction j; P_i and P_{i+1} are the total pressures in volumes i and i+1 across junction j, including gravity head.

3.6.5 PROTECTIVE MEASURES

Piping within the plant is arranged or restrained such that in the event of a LOCA, the dynamic effects associated with the pipe rupture will not result in loss of containment integrity or prevent engineered safety features from mitigating the effects of the LOCA.

The containment vessel is protected from the effects of LOCA pipe rupture by the secondary shield wall. The secondary shield wall encloses all piping whose failure could cause a LOCA. Pipe whip restraints are provided on all such lines which are connected to containment penetrations to limit the pipe rupture loads on the penetrations.

Protection of engineered safety features against the effects of LOCA pipe ruptures is provided by any or a combination of the following:

a) Spatial separation of redundant pipings and components such that pipe whip or jet impingement resulting from the LOCA cannot damage both redundant components.

b) Placing of pipe whip restraints such that resulting pipe movement cannot damage adjacent piping or components.

c) Placing of barriers between redundant components.

Where required, protection against damage to adjacent piping or equipment from pipe whip is provided by limiting the amount of movement of the ruptured pipe. This is achieved by placing pipe whip restraints such that the reaction forces on the broken pipe do not cause formation of a plastic hinge.

Restraint spacings have been developed for rupture in all typical piping configurations as shown on Figure 3.6-1. Reaction forces ire calculated using the relationships in Section 3.6.4.

The design loads for pipe whip restraints are determined by the location of the pipe rupture in relation to the physical location of the pipe restraint. To assure conservatism, pipe ruptures are postulated in a location on the system which would create the largest forces and moments on the restraints. The magnitude of the load (static) corresponds to the pipe reaction or jet impingement forces.

Since this load is applied suddenly, the restraints are designed for impact loading. This is accomplished by increasing the static nature of the load by a dynamic load factor and the restraints and restraint structures are analyzed in terns of an equivalent static load.

The magnitude of the dynamic load factor varies depending, on the rapidity and inelasticity of the load application and the dampening characteristics of the system. It is known, however, that the maximum value, based on a single degree of freedom system is 2.0. To assure conservatism, a design load factor of 2.0 has been used for design of pipe whip restraints.

Where required, protection against damage to piping or equipment from jet impingement from an adjacent ruptured pipe is provided by ensuring that the spacing between piping or equipment is such that the jet impingement loads do not result in failure of the piping or equipment being impinged upon. the jet impingement forces are calculated using the relationships given in Section 3.6.4. The impingement force is attenuated with distance from the break according to the variation of jet velocity and divergence with distance.

There is no critical equipment exposed to a reactor coolant pipe break blowdown. Critical piping underwent analysis to assure that there would be no failures as a result of the LOCA jet impingement. The critical piping system pipe break forces were compared to those that could result from the jet impingement (this includes the cantilever effect on the nozzles where the critical piping penetrates the vessel) to determine which was larger. Additional restraints were added where the jet impingement forces were larger.

3.6.5.1 Spacing of Pipe Restraints

a) Design approach

For the pipe rupture loading condition, the design philosophy is that the ruptured pipe itself must be restrained in such a way that it does not develop pipe whipping and consequently impair a nearby pipe, critical structure or pieces of equipment.

To implement the above, the load carrying capacities of the pipe under the postulated "maximum credible load evaluated in terms of its ultimate capabilities to resist torsion and/or bending;i.e. the pipe is allowed to experience permanent deformation without loss of function of the system. once the ultimate load carrying capabilities of the pipe are established, the spacing of the pipe restraints can then be designed to prevent pipe whipping.

b) Ultimate Load Capability

In the evaluation of the ultimate load capability, it is assumed that a structure fails when the applied loads produce maximum primary stresses equal to those corresponding to 50 percent of the total strain on the trapezoidal stress-strain curve as shown in Figure 3.6-9A. This consideration accounts for the difference between actual and theoretical structures due to the presence of welding or connections. The basis for the above assumption is the work performed at the United States Naval Ordnance Laboratory. The Naval Lab work shows that models of vessels with welding and connections burst when the maximum strain was of the order of 50 percent of the total strain.

To simplify the analysis, the stress-strain diagram is approximated by a trapezoidal curve, as shown in Figure 3.6-9A. The design failure stress, S_u*. material is obtained from the resulting diagrams at the strain ε_u*. = 0.50 ε_u.

This linearized approach introduces further conservatism because the failure stress thus selected corresponds to a strain less than 50 percent of the total strain on the actual stress-strain diagrams. For ferritic material with a pronounced yield point this difference is small, but significant. For stainless steel, the difference is greater. The slope of the actual stress-strain diagram is much steeper than the linear model used in the calculations of the failure stresses in the region of the yield point indicating that most of the strain hardening resulting in increased strength occurs as soon as the structure starts to yield.

Structures having hollow-circuit cross sections are considered approximating nozzles or pipes carrying steam or water under pressure, when subjected to mechanical loads across their total cross section. The method used by Stokey, Peterson and Wunder [7] to evaluate limit loads for tubes under internal pressure, bending moment, axial .force and torsion rigid-plastic material without strain-hardening has been adopted and amplified to account for the effect of strain hardening. This effect has been evaluated following the method outlined in References (8), (9), and (10). The Tresca, or maximum shear stress, theory has been applied in this development.

c) Analytical Methods

The principal stresses in a Mohr's circle are

$$S_1 = \frac{S_m + S_t}{2} + \sqrt{\left(\frac{S_t - S_m}{2}\right)^2 + S_s^2} \qquad (1)$$

$$S_2 = \frac{S_m + S_t}{2} - \sqrt{\left(\frac{S_t - S_m}{2}\right)^2 + S_s^2} \qquad (2)$$

$$S_3 = -\frac{P}{2} \qquad (3)$$

S_m = circumferential stress

S_t = axial tensile stress

S_s = shear stress

P = Internal pressure

In the case of $S_2 > S_3$, the stress intensity, i.e., S.I. is given by:

$$S.I. = S_1 - S_3 = \frac{S_m + S_t}{2} + \sqrt{\left(\frac{S_t - S_m}{2}\right)^2 + S_s^2} + \frac{1}{2}p \qquad (4)$$

In the case of $S_2 < S_3$:

$$S.I. = S_1 - S_2 = 2\sqrt{\left(\frac{S_t - S_m}{2}\right)^2 + S_s^2} \qquad (5)$$

Using the maximum shear stress criterion, the limit condition is:

$$S.E. = S_u^* \qquad (6)$$

that is, when $S_2 > S_3$:

$$1/2\,(S_t + S_m) + \sqrt{\left[\frac{1}{2}(S_t - S_m)\right]^2 + S_s^2} + \frac{1}{2}p = S_u^* \qquad (7)$$

and, when $S_2 < S_3$:

$$2\sqrt{\left(\frac{S_t - S_m}{2}\right)^2 + S_s^{\ 2}} = S_u^* \qquad (8)$$

Equations (7) and (8) yield the lower limits on the allowable axial tensile stress. Rewriting the above equations:

$$\left(\frac{S_t}{S_y}\right)_{all.} = K_1\left[\frac{S^2 - S\left(S_m/K_1S_y\right) - \left(S_s/K_1S_y\right)^2}{S - \left(S_m/K_1S_y\right)}\right] \text{ for } S_2 > S_3 \qquad (9)$$

$$\left(\frac{S_t}{S_y}\right)_{all.} = K_1\left[\frac{S_m}{K_1S_y} + \sqrt{1 - 4\left(\frac{S_s}{K_1S_y}\right)^2}\right] \text{ for } S_2 < S_3 \qquad (10)$$

where:

$$S = 1 - P/2K_1 S_y \qquad (11)$$

$$K_1 = S_u^*/S_y = 1 + (S_u/S_y - 1)\ \varepsilon_u^* / \varepsilon_u \qquad (12)$$

For the case of no shear stress, Equations (9) and (10) give:

$$\left(\frac{S_t}{S_y}\right)_{all.} = K_1\left[1 - \frac{P}{2K_1S_y}\right] \qquad (13)$$

$$\left(\frac{S_t}{S_y}\right)_{all.} = K_1\left[1 + \frac{S_m}{K_1S_y}\right] \qquad (14)$$

For the combinations of stresses present in the components of piping systems, S_z is usually greater than S_3, therefore the equations that normally apply are (4), (7), (9) and (13).

For the compressive stress region, the previously mentioned cases, i.e., $S_2 > S_3$ and $S_2 < S_3$, also occur. In the components of piping systems S_2 is algebraically smaller than S_3.

Using the maximum shear stress (τ_m) criterion, the limit condition is:

$$\tau_m = \sqrt{\left[\frac{1}{2}\left(S_m - S_c\right)\right]^2 + S_s^{\,2}} = \frac{1}{2} S_u^* \tag{15}$$

Where S_c = axial compressive stress. This gives:

$$\left(\frac{S_c}{S_y}\right)_{all.} = K_1\left[S_m/K_1S_y - \sqrt{1 - 4\left(S_s/K_1S_y\right)^2}\right] \tag{16}$$

For the case of no shear stress, equation (16) becomes:

$$\left(\frac{S_c}{S_y}\right)_{all.} = -K_1\left(1 - \frac{S_m}{K_1S_y}\right) \tag{17}$$

The axial load N, and the bending moment, M can then be written in terms of $(S_c)_{all.}$ and $(S_t)_{all}$. Due to the presence of stresses in different planes, the compressive and tensile stress distributions do not necessarily have to be symmetrical.

The different combinations of N and M that are assumed to cause "collapses" are given by the trapezoidal stress distributions with the maximum tensile and/or compressive axial stresses as calculated from equations (9) and (16).

Figure 3.6-9B shows a typical stress distribution when the pipe begins to collapse. In this figure, it has been assumed that the extreme fiber stress in compression reaches $(Sc)_{all.}$ and $S_y \le S_t \le (S_c)_{all}$.

Another possible case, although not depicted here, is that the extreme fiber stress in tension reaches $(S_t)_{all.}$ and $S_y < \left| S_c \right| < (S_t)_{all}$.

However, since most common steels display greater strength in tension that in compression, the stress growth is faster in the compressive region. It is, therefore, reasonable to disqualify the latter in favor of the former stress distribution.

The axial force, N, and the bending moment, M, are defined as follows:

$$N = \int_A S \, dA \tag{18}$$

$$M = \int_A S \, y \, dA \tag{19}$$

It has been shown that ultimate load capabilities for pipe can very well be based upon assuming thin walled tubing. Introducing cylindrical coordinates and assuming a thin-walled tube, the following expressions can be obtained:

$$y = r \sin \psi \tag{20}$$

$$dA = t\, r\, d\psi \tag{21}$$

For the example illustrated in Figure 3.6-9B, the axial force, N, and the bending moment, M, are then given by:

$$N = 2\,tr\left(\int_{\theta}^{\frac{\pi}{2}} S_t \, d\Psi + \int_{-\frac{\pi}{2}}^{\theta} S_c \, d\Psi\right) \tag{22}$$

$$M = 2\,tr^2\left(\int_{\theta}^{\frac{\pi}{2}} S_t \sin\Psi \, d\Psi \int_{-\frac{\pi}{2}}^{\theta} S_c \sin\Psi \, d\Psi\right) \tag{23}$$

By proportioning of the stress distribution diagram in Figure 3.6-9B and S_c can be expressed as

$$S_t = S_y + \left(S_t' - S_y\right) \frac{\operatorname{Sin}\psi - \operatorname{Sin}\theta}{1 - \operatorname{Sin}\theta} \tag{24}$$

$$S_t = -S_y + \left(\left|\left(S_c\right)_{all}\right| - S_y\right) \frac{\operatorname{Sin}\psi - \operatorname{Sin}\theta}{1 + \operatorname{Sin}\theta} \tag{25}$$

The bending moment increases as the angle, θ, approaches zero.

By inspection of equations (9) and (16), it is found that $(S_t)_{all.}$

is always greater than $\left|(S_c)_{all}\right|$ Since S_t' is less than $(S_t)_{all}$, conceivably S_t can equal $(S_c)_{all}$

The axial force, N, is calculated from equations (22), (24) and (25) and it is consequently assumed to be zero. The effect of axial force on the maximum moment capability is negligible for reasonable pressures and standard pipe sizes (7). Also a normally restrained pipe run does not exhibit any axial loads under the considered loading conditions.

Thus

$$N = 2\,tr\left\{-2Sy\theta + \cos\theta\left(\frac{St' - S_y}{1-\sin\theta} - \frac{\left|(S_c)_{all}\right| - S_y}{1+\sin\theta}\right) + \right.$$

$$\left. \mathrm{Sin}\theta\left(\frac{St' - S_y}{1-\sin\theta}\left[\theta - \frac{\pi}{2}\right] - \frac{\left|(S_c)_{all}\right| - S_y}{1+\sin\theta}(\theta + \frac{\pi}{2})\right)\right\} = 0 \tag{26}$$

By inspection of equation (26), it is clear that

$$S_t' = \left|(S_c)_{all}\right| \quad \text{when } \theta = 0 \tag{27}$$

This condition gives the ultimate moment which can be evaluated from equations (23), (24), (25), and (27). Thus,

$$M = 4\,tr^2\,S_y\left\{1 + \frac{\pi}{4}\left(\left|\left(\frac{S_c}{S_y}\right)_{all}\right| - 1\right)\right\} \tag{28}$$

By substituting equation (12) into equation (16), and making use of the postulated assumption that $\varepsilon_u{}^*/\varepsilon_u = 1/2$, the following is true:

$$\left|\left(\frac{S_c}{S_y}\right)_{all}\right| = \frac{1}{2}\sqrt{\left(1+\frac{S_u}{S_y}\right)^2 - \left(4\frac{S_s}{S_y}\right)^2} - \frac{S_m}{S_y} \tag{29}$$

Equations (28) and (29) give the dimensionless ultimate moment as

$$\frac{M}{M_o} = 1 + \frac{\pi}{8}\left[\sqrt{\left(1+\frac{S_u}{S_y}\right)^2 - \left(4\frac{S_s}{S_y}\right)^2} - 2\left(\frac{S_m}{S_y} + 1\right)\right] \tag{30}$$

where $M_o = 4\,tr^2 S_y$ (31)

The ultimate torque-moment for a thin-walled tube can be expressed in terms of the shear stress, S_s as

$$T = 2\pi r^2\,t\,S_s \tag{32}$$

or in dimensionless form

$$\frac{T}{T_o} = \frac{S_s}{S_y} \qquad (33)$$

where $To = 2\pi r^2 t S_y$ (34)

Substituting equation (33) into equation (30) yields:

$$\frac{M}{M_o} = 1 + \frac{\pi}{8}\left[\sqrt{\left(1+\frac{S_u}{S_y}\right)^2 - \left(4\frac{T}{T_o}\right)^2} - 2\left(\frac{S_m}{S_y}+1\right)\right] \qquad (35)$$

d) Maximum Spacing of Constraints-for Pipe Rupture Loading

Generally, the severity of the guillotine type rupture (circumferential rupture) by far exceeds the effect of the slot type failure regardless of piping configuration. It is, therefore, sufficient to utilize the former loading condition to determine the maximum allowable unrestrained length of pipe run. Since most piping layouts are designed with straight runs and 90° bends or elbows, interaction between bending and torsion seldom occurs. Hence moment calculations due to external loadings are greatly simplified (see preceding subsection C). The moment is simply calculated as the force times the perpendicular distance to the point of constraint or restraint.

Assuming that a broken pipe can be represented as a cantilever,
the moment under bending at the constraint for restraint is given by:

$$M = k\,PAL \qquad (36)$$

where kPA is the force at the break, and L is the distance from the break to the restraint, and

$$A = \pi\left(\frac{D_o - 2t}{w}\right)^2$$

The reduced moment can now be calculated from equations (31) and (36).

$$\frac{M}{M_o} = \frac{kP\pi L\,[D_o - 2t\,]^2}{4t\,S_y\,[D_o - t\,]^2} \qquad (37)$$

To prevent further pipe rupture or excessive deflection, this reduced moment must be less than the expression per equation (35) with $T/T_o = 0$

Thus,

$$\frac{\pi k PL(D_o - 2t)^2}{4t S_y (D_o - t)^2} < 1 + \frac{\pi}{8}\left[\frac{S_u - S_y}{S_y} - P\frac{(D_o - t)}{tS_y}\right] \qquad (38)$$

Introducing the dimensionless variables $\eta = t / D_o$

$\Phi = \pi P / S_y + (\pi/8)(S_u - S_y)$, this can be written as

$$\frac{kL}{D_o} < \frac{4\eta(1-\eta)^2}{(1-2\eta)^2}\left[\frac{1}{\varphi} - \frac{1}{8}\left(\frac{1-\eta}{\eta}\right)\right] \qquad (39)$$

Under twisting of the pipe, the dimensionless unrestrained length kL/D_o is similarly expressed for the case of torsion as:

$$\frac{kL}{D_o} < \frac{\eta(1-\eta)^2}{(1-2\eta)^2}\sqrt{\left[\frac{(S_y + S_u)^2}{(2P)^2} - \left(\frac{1-\eta}{2\eta} - \frac{S_y}{P}\left(1 - \frac{4}{\pi}\right)\right)^2\right]} \qquad (40)$$

If the above inequalities are satisfied, then span length selected is considered acceptable.

e) Primary System

Circumferential (guillotine) and longitudinal (slot) breaks were postulated for the RCS hot and cold legs in the original plant design. Since then, however, the revision to 10 CFR 50 General Design Criteria (GDC) 4 allows elimination of the consideration of dynamic effects of these loss of coolant accidents from the plant design bases. The dynamic effects of a LOCA include the effects of missiles, pipe whipping, discharging fluid (i.e., jet impingement), decompression waves within the ruptured pipe and dynamic or nonstatic pressurization in cavities, compartments, and subcompartments. Reference 24 demonstrates that the primary loop piping meets all of the criteria for application of leak before break presented in NUREG-1061, Volume 3. As a result, the mechanical/structural loads associated with dynamic effects of guillotine and slot breaks in RCS hot and cold legs are no longer considered a plant design basis (References 26 and 27).

The original pipe rupture criteria used to design the primary coolant loop restraints is described below for historical purposes. It is as follows:

a) The supports for the steam generators and reactor are designed to accommodate pipe rupture loadings associated with a LOCA (see Section 3.9.1.4.2)

b) The reactor coolant pumps are restrained from becoming a missile in the event of a LOCA.

c) The primary coolant piping in the primary shield wall are restrained to minimize pipe separation in the event of a guillotine rupture of either the hot or cold leg.

The location of the piping restraints in accordance with criterion b) and c) are shown on Figure 3.6-50. The restraints to meet criterion b) include reactor coolant pump suction line stops, restraints on the RC pump casing, and around the RC pump motor. The restraints to meet criterion c) include the stops around the primary coolant loops in the primary shield wall and the wall itself. A guillotine of the cold leg at the nozzle of the steam generator is the only leg expected to form a plastic hinge. This is confirmed by the analysis described above. The force will cause the pipe to be driven to the floor.

There is no critical equipment exposed to a LOCA blowdown. critical piping underwent analysis to assure that there would be no failures as a result of the LOCA jet impingement. The critical piping system pipe break forces were compared to those that could result from the jet impingement (this includes the cantilever effect on the nozzles where the critical piping penetrates the vessel) to determine which was larger. Additional restraints were added where the jet impingement forces were larger.

REFERENCES FOR SECTION 3.6

1. AEC letter to Florida Power & Light Co., Enclosure No. 3, dated 8/30/73

2. R. E. Keever and G. A. Randell, "Analysis for Effects of Pipe Rupture - State of the Art," - Paper presented at ANS Summer Meeting, Chicago, Illinois, June 1973

3. IDO - 17242 (May 1968), IDO - 17266 (Feb. 1969), USAEC National Reactor Testing Station Quarterly Technical Records

4. "The Theory of Turbulent Jets," N. G. Abramovitch, MIT Press, 1963

5. US Naval Ordance Laboratory, NOLTR 63-140 - Explosion Containment Laws for Nuclear Reactor Vessels - Section 7.4, 16 August 1965, W. R. Wise, Jr., and J. R. Proctor.

6. USAEC Docket 50-247, Exhibit B-2 Indian Point Unit No. 2, Second Supplement, Appendix B.

7. Limit Loads for Tubes Under Internal Pressure Bending Moment, Axial Force and Torsion, W. F. Stokey, D. B. Peterson and R. A. Wunder - Nuclear Engineering and Design 4 (1966) 193-201. North-Holland Publishing Company, Amsterdam.

8. "Introduction to Plasticity," Aris Phillips - The Ronald Press Company - New York, 1956.

9. Scienza delle Construzioni, Odone Belluzzi, Volume 3, Zanichelli Editore - Bologna, 1965.

10. "Aircraft Structures," D. J. Peery - McGraw-Hill Book Company, Inc., 1950.

11. "Flow of Fluids Through Valves, Fittings, and Pipes," Crane Co., Chicago, Illinois, 1942.

12. F. J. Moody, "Maximum Flow Rate of a Single Component Two Phase Mixture," Journal of Heat Transfer, Trans. ASME, Series C, Vol. 87, Feb. 1965.

13. Haunsaker, J. C. and Rightmire, B.G., "Engineering Applications of Fluid Mechanics," McGraw-Hill, 1947.

14. Rettig, W. H. et al, "RELAP-3 - A Computer Program for Reactor Blowdown Analysis," IN-1321, June 1970.

15. Abramovich, "The Theory of Turbulent Jets," MIT Press, 1960.

16. 'Dynamics of Structures' - Hurty, W. C., Rubenstein, M.F. pp. 255, 275; Prentice Hall 1964.

17. 'The Mathematic Theory of Plasticity'-Hill, R. Oxford Univ. Press, 1950.

18. 'Plasticity: Theory and Application' - Mendelson, A. MacMillan 1968.

19. 'A General Procedure for the Analysis of Elastic and Plastic Frameworks'-Morris, G. A., Fenves, S. J.

20. 'Yielding in Framed Structures Under Dynamic Loads'-Nigam, N.C. J. Eng. Mech. Proc. ASCE, Oct 1970, EM5, pp. 687, 709.

21. 'A Method of Computation for Structural Dynamics'-Newmark, N. M. J. Eng. Mech. Proc. ASCE, July 1959, EM3, pp. 67, 94.

22. Letter R. E. Uhrig (FPL) to R. A. Clark (NRC) Re: St. Lucie Unit 1, Docket 50-335, Asymmetric LOCA Loads, L-80-263 dated 8/8/80.

23. Letter J. A. Norris, Sr., (NRC) to C. O. Woody (FPL) Re: Asymmetric LOCA Loads - St. Lucie Plant, Unit 1 (TAC No. 08923), dated August 9, 1989.

24. CE Owners Group Report CEN-367-A, Leak-Before-Break Evaluation of Primary Coolant Loop piping in Combustion Engineering Designed Nuclear Steam Supply Systems.

25. Federal Register, Volume 52, dated October 27, 1987.

26. Letter D.A. Sager (FPL) to NRC, “St. Lucie Units 1 and 2 Application of Leak-Before-Break Technology to Reactor Coolant System Piping,” L-92-231 dated 8/26/92 and Letter J.A. Norris, Sr., (NRC) to J.H. Goldberg, “St. Lucie Units 1 and 2 - Application of Leak-Before-Break Technology to Reactor Coolant System Piping,” dated March 5, 1993.

27. NRC Generic Letter 87-11, Subject: Relaxation and Arbitrary Pipe Rupture Requirements, Issued June 19, 1987.

28. USNRC Standard Review Plan, Section 3.6.2, Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping, Revision 2, June 1987.

TABLE 3.6-1

SENSITIVITY STUDY FOR 20" FW LINE OUTSIDE CONTAINMENT

Node No. At Restraint	Gap (in)	Wall Thick. (in)	Max. Reaction (lbs)	Max. Plastic Strain in Pipe	Max. Defl. At Loaded Node
5	2	1	$.8721 \times 10^6$	.0065	6.381"
5	2	1.5	$.68 \times 10^6$	.0072	5.446"
5	4	1.5	$.7806 \times 10^6$	.00148	9.255"

TABLE 3.6-2

SUMMARY OF PIPE WHIP ANALYSIS

Line Description	Guillotine Break at Node Number	Active Restraint Node Number	Maximum Deflection at break	Maximum Reaction at Restraint (pounds x 10^6)	Maximum Plastic Strain in Pipe
Feedwater Outside containment 1.5" wall and 4.00" gap	7	5	9.255"	0.7806	0.00148
Feedwater Inside Containment	7	6	3.502"	1.34	0.00259
Mainsteam Inside Containment	12	9	11.46"	2.72* (+X) 2.34 (+Z)	0.00762
Mainsteam Inside Containment	16	9 13	13.36"	1.0 (+Z) 1.78 (+X) 1.52 (+X,-Z)	0.0000

*Values exceeding 2KPA have a duration of <2 milliseconds

TABLE 3.6-3

MAIN STEAM LINE INSIDE THE CONTAINMENT
GUILLOTINE BREAK AT NODE #12

Node	Restraint I.D. No,	El. Spring Const.	Pl. Spring Const.	Gap (in.)
2	MS-12	8.536×10^{6}#/in	3.570×10^{5} #/in	6.00
4	MS-13	8.786×10^{6}#/in	3.650×10^{5}#/in	6.00
8	MS-15	8.786×10^{6}#/in	3.650×10^{5}#/in	5.50
9	MS-16	8.786×10^{6}#/in	3.650×10^{5}#/in	5.50
11	MS-17	8.786×10^{6}#/in	3.650×10^{5}#/in	4.00

TABLE 3.6-4

MAIN STEAM LINE INSIDE THE CONTAINMENT
GUILLOTINE BREAK AT NODE #16

Restraint Information

Node	Restraint I.D. No.	El. Spring Const.	Pl. Spring Const.	Gap (in.)
2	MS-12	8.536×10^{6} #/in	3.570×10^{6} #/in	6.00
4	MS-13	8.786×10^{6} #/in	3.650×10^{6} #/in	6.00
8	MS-15	8.786×10^{6} #/in	3.650×10^{6} #/in	5.50
9	MS-16	8.786×10^{6} #/in	3.650×10^{6} #/in	5.50
11	MS-17	8.786×10^{6} #/in	3.650×10^{6} #/in	4.00
13	MS-18	8.786×10^{6} #/in	3.650×10^{6} #/in	4.00

TABLE 3.6-5

BOILER FEEDWATER LINE OUTSIDE THE CONTAINMENT
GUILLOTINE BREAK AT NODE #7

Restraint Information

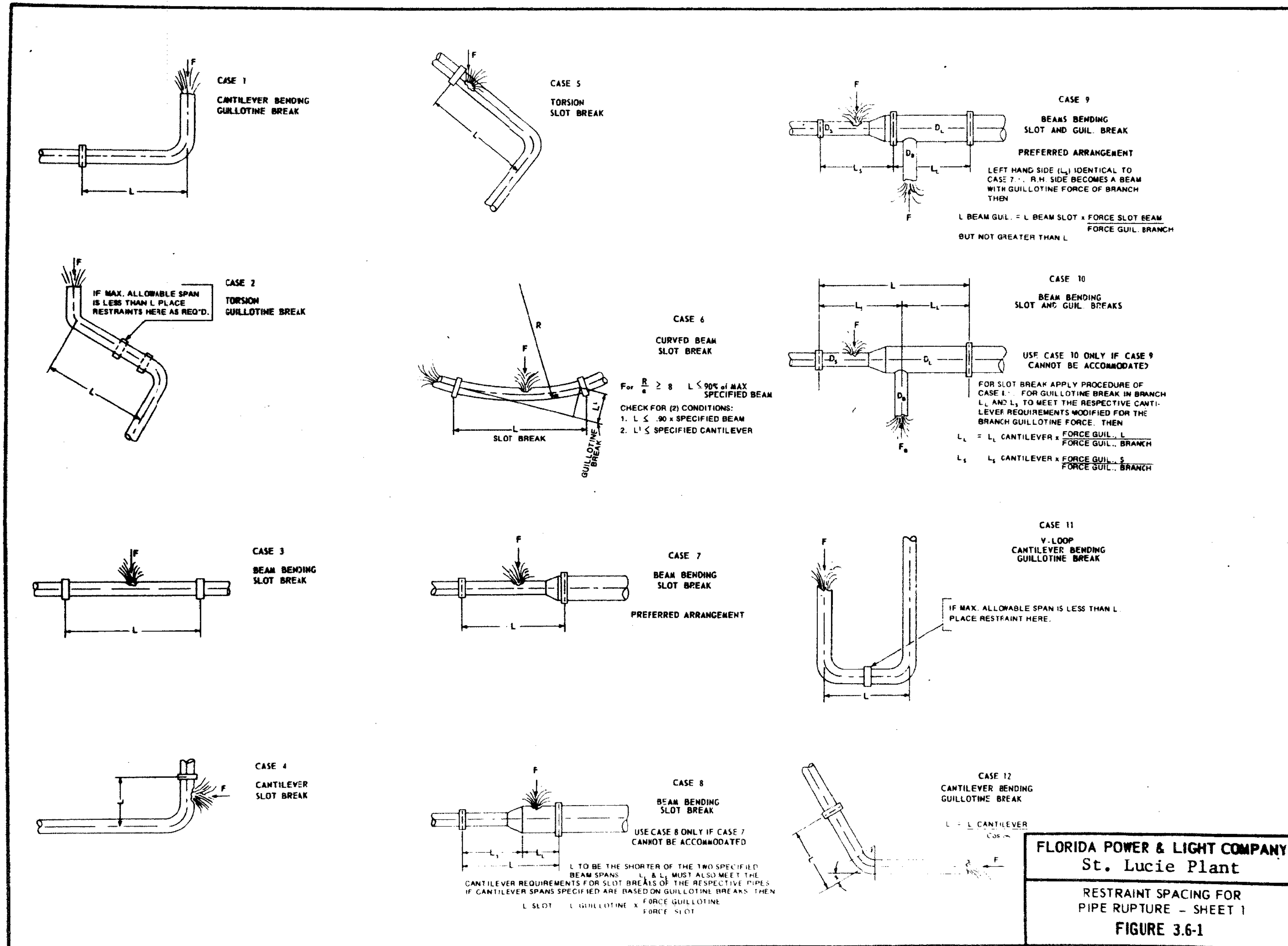
Node	El. Spring Const.	Pl. Spring Const.	Gap (in.)
2	8.536 x 10^6 #/in	3.570 x 10^6 #/in	4.00
4	8.536 x 10^6 #/in	3.570 x 10^6 #/in	4.00
5	8.536 x 10^6 #/in	3.570 x 10^6 #/in	4.00

TABLE 3.6-6

BOILER FEEDWATER LINE INSIDE THE CONTAINMENT
GUILLOTINE BREAK AT NODE #7

Restraint Information

Node	El. Spring Const.	Pl. Spring Const.	Gap (in.)
4	8.786 x 10^6 #/in	3.650 x 10^6 #/in	2.50
6	8.786 x 10^6 #/in	3.650 x 10^6 #/in	4.00



FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESTRAINT SPACING FOR PIPE RUPTURE - SHEET 1

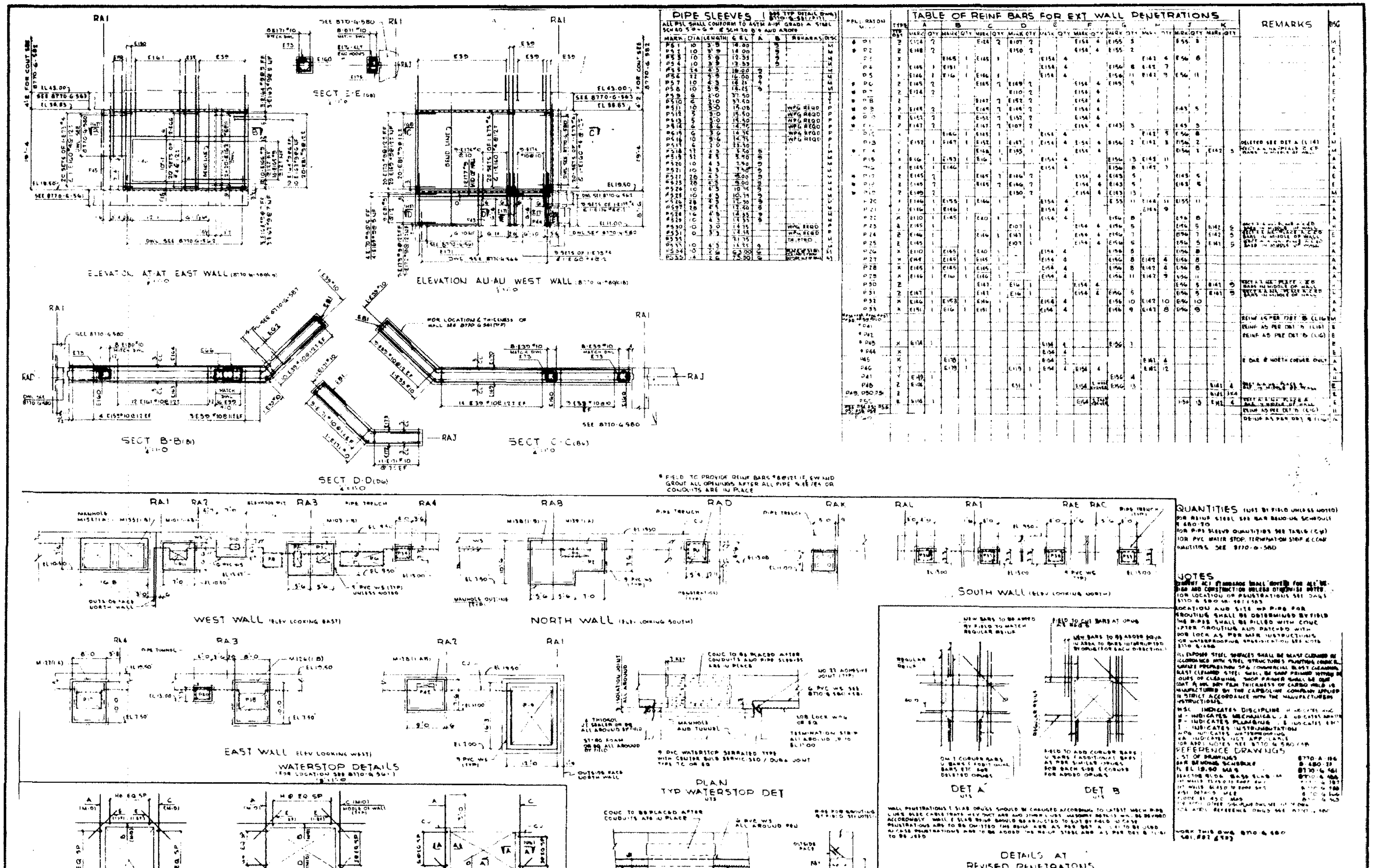
FIGURE 3.6-1

Refer to drawing
8770-G-799 Sheet 13

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING PIPE RESTRAINTS SH.13

FIGURE 3.6-2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

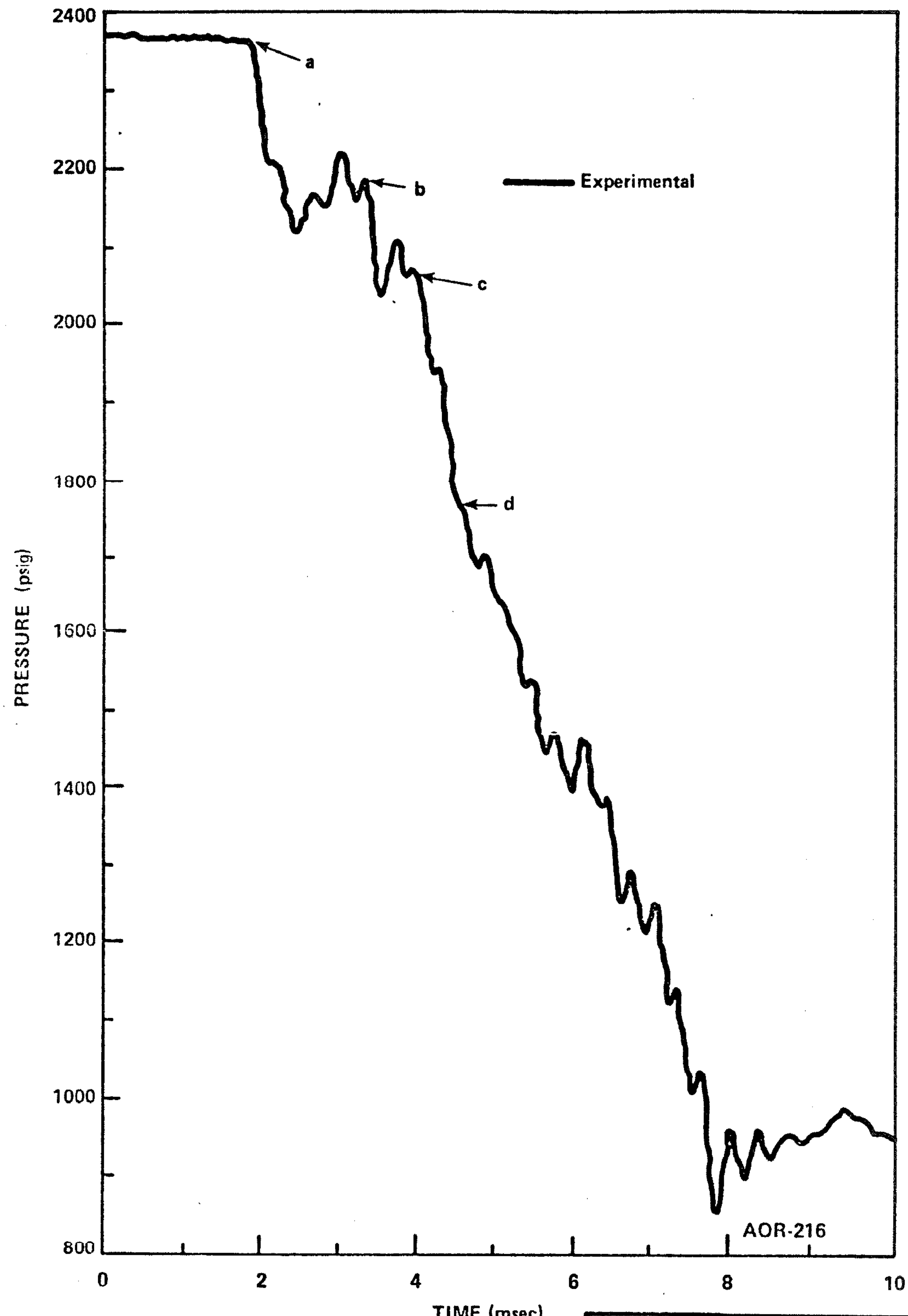
CONFIGURATION AND DETAILS FOR PIPE WHIP RESTRAINTS
TYPICAL
FIGURE 3.6-3

DELETED

Amendment No. 23 (11/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

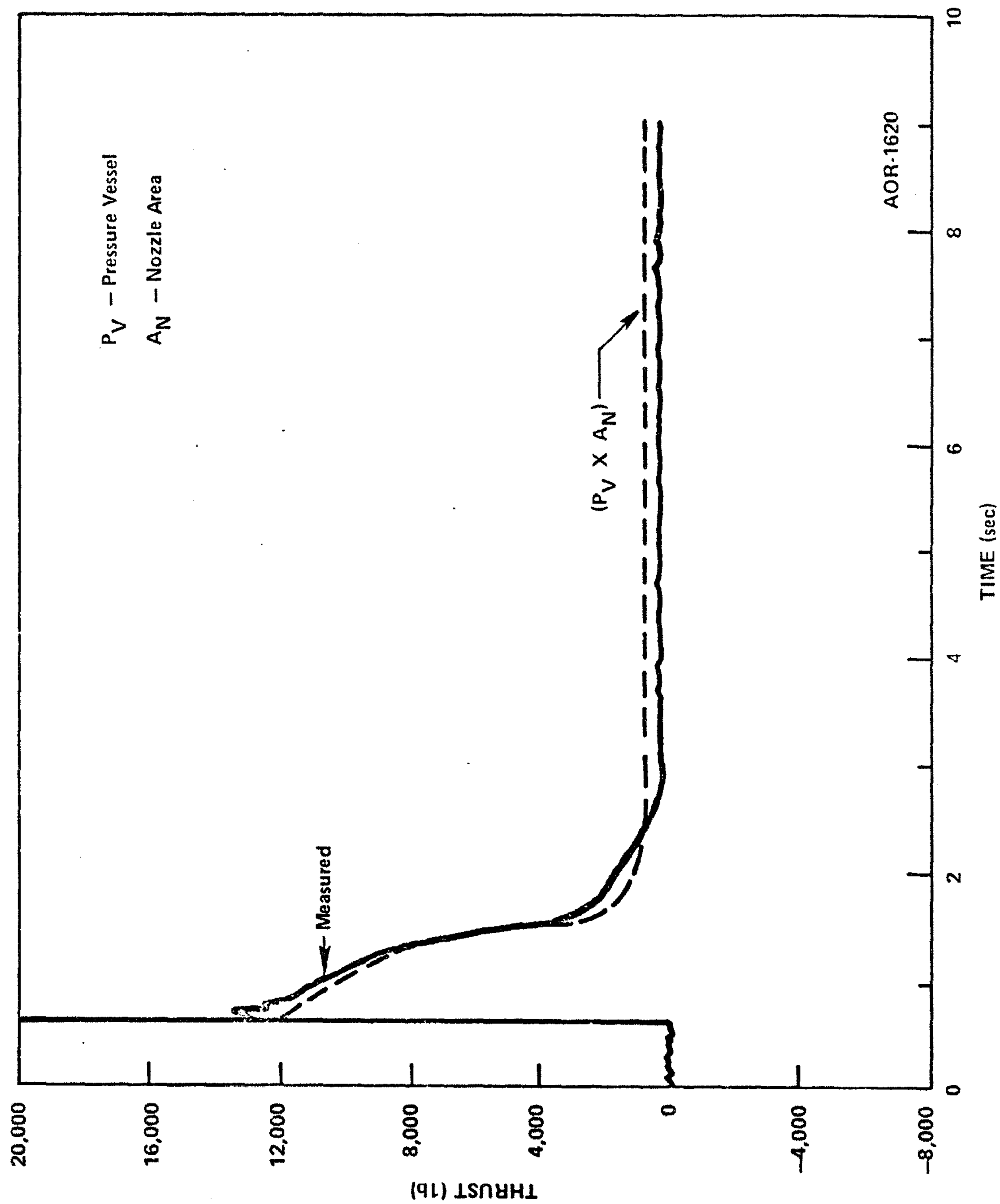
FIGURE 3.6-4



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

PIPE RUPTURE
PRESSURE HISTORY

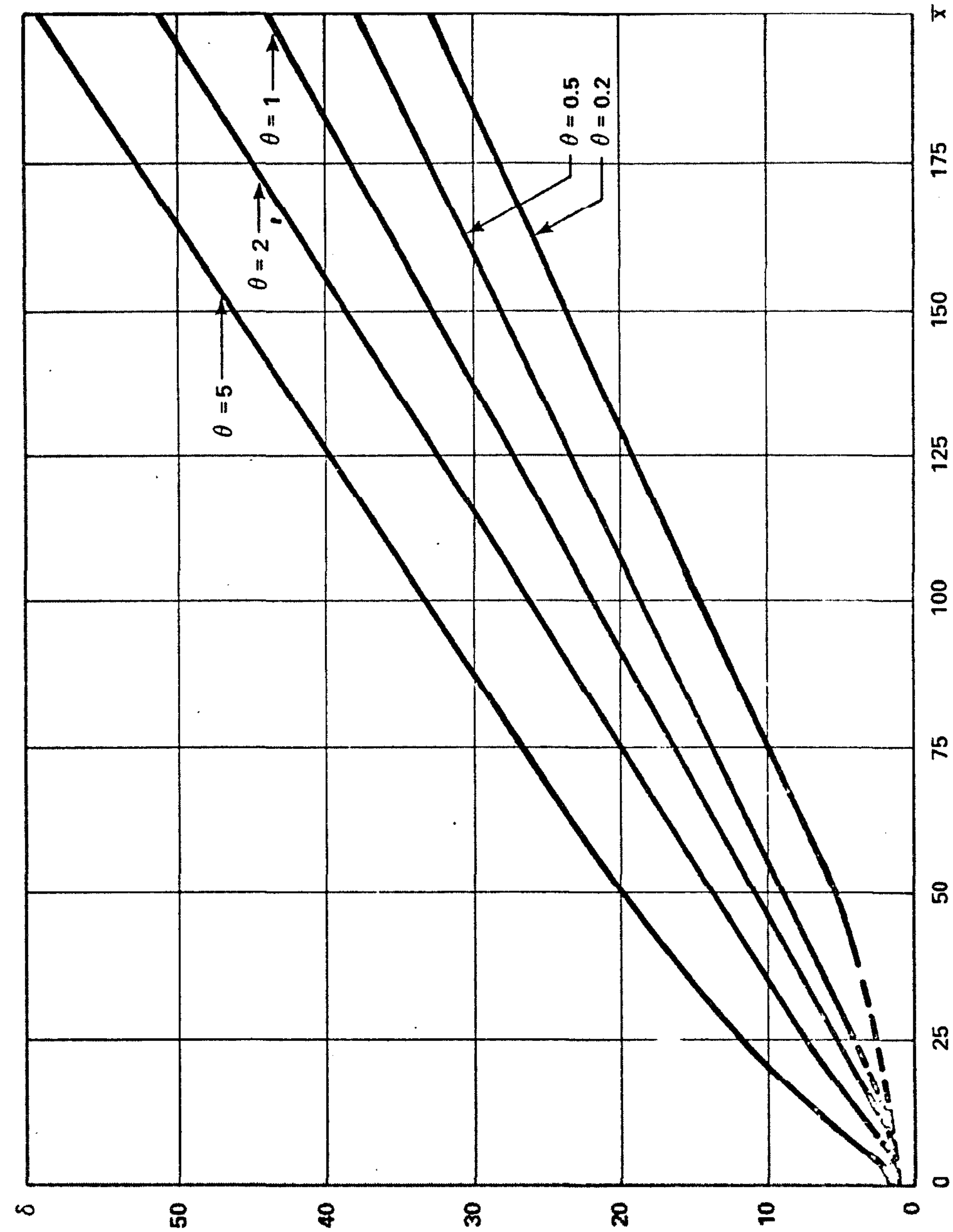
FIGURE 3.6-5



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

COMPARISON OF MEASURED AND CALCULATED THRUSTS

FIGURE 3.6-6

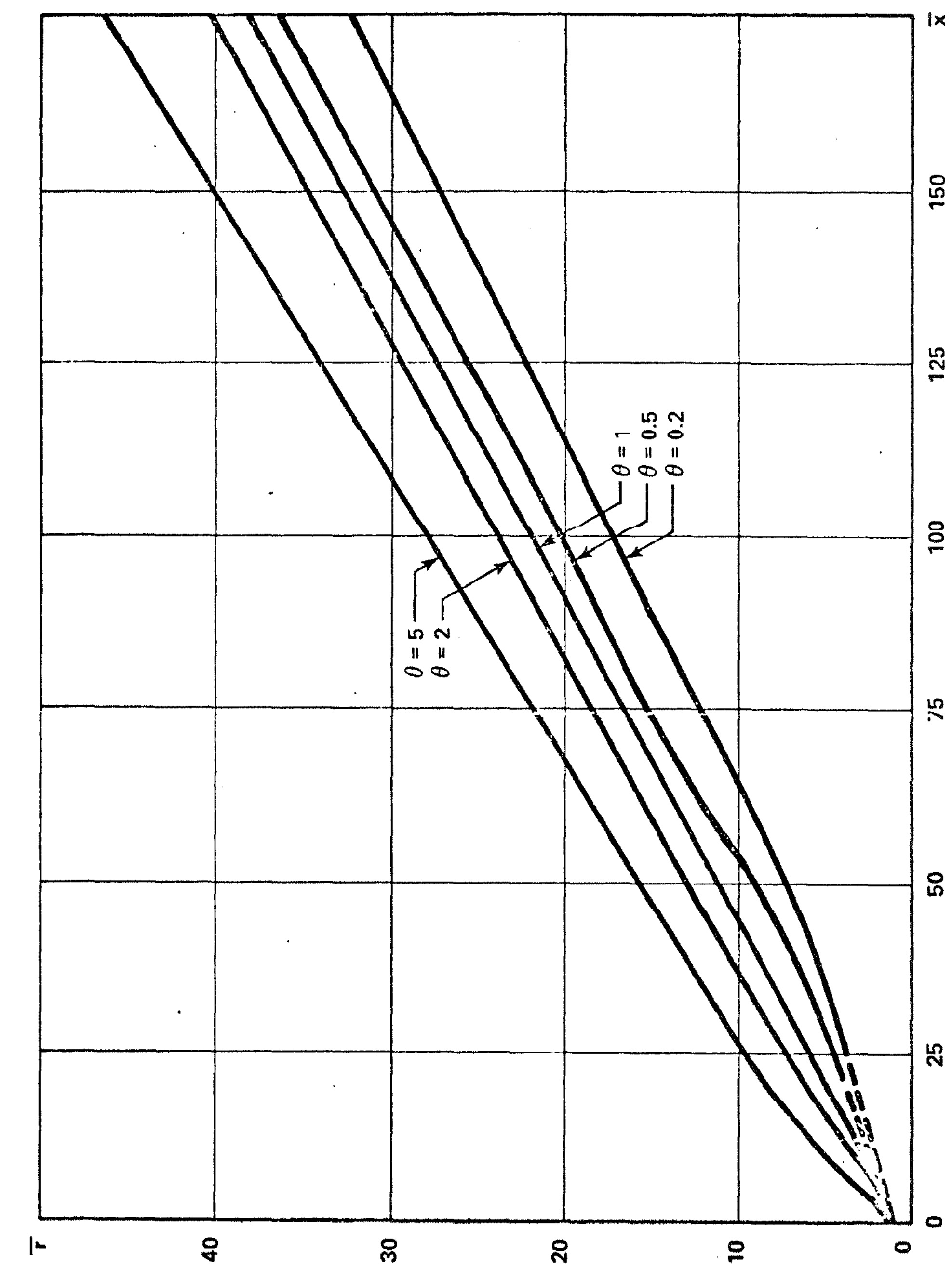


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

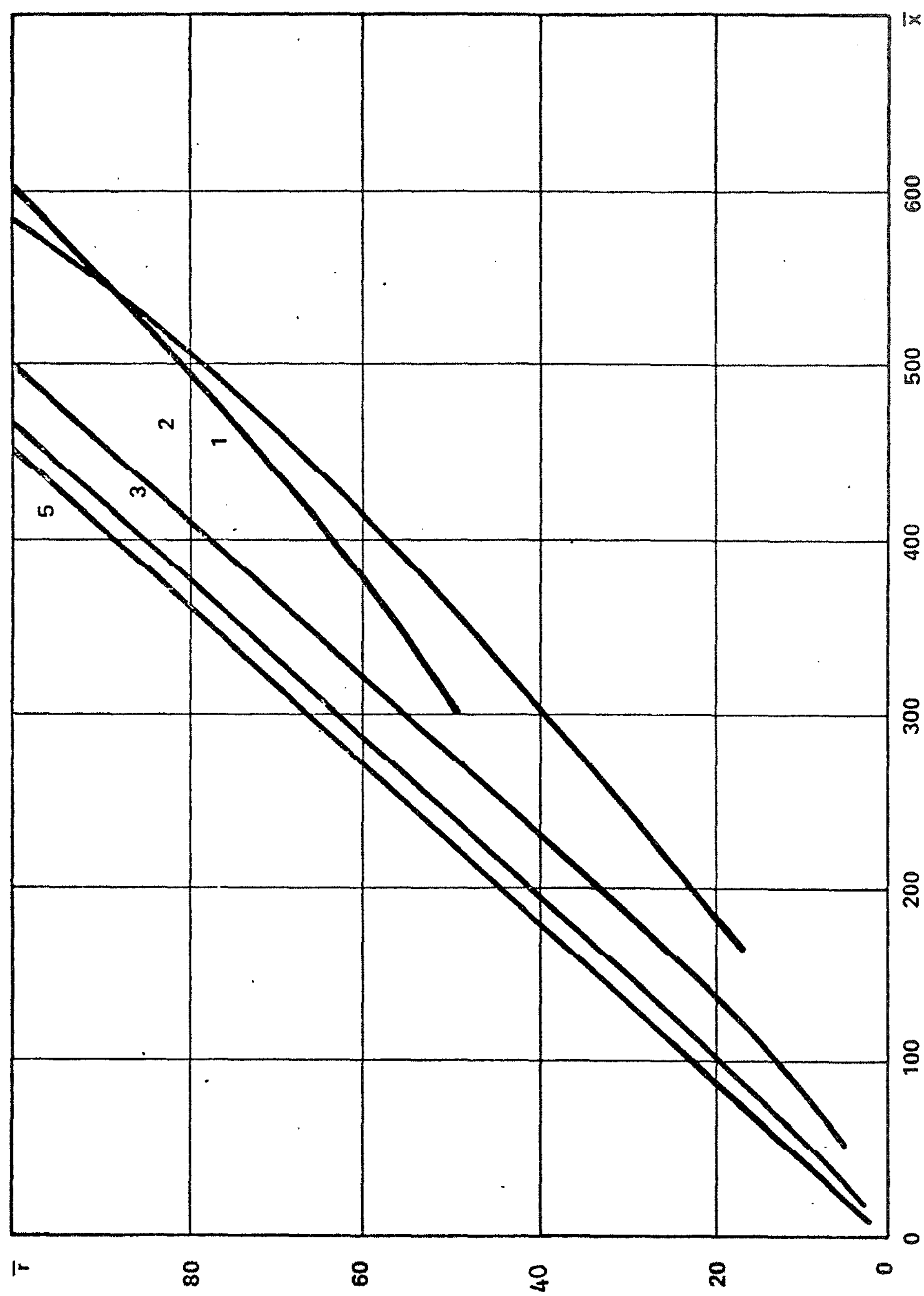
GROWTH ALONG THE AXIS OF THE THICKNESS OF A NONISOTHERMAL TWO-DIMENSIONAL GAS JET IN A STATIONARY MEDIUM, (INFLUENCE OF PARAMETER θ).

FIGURE 3.6-7



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

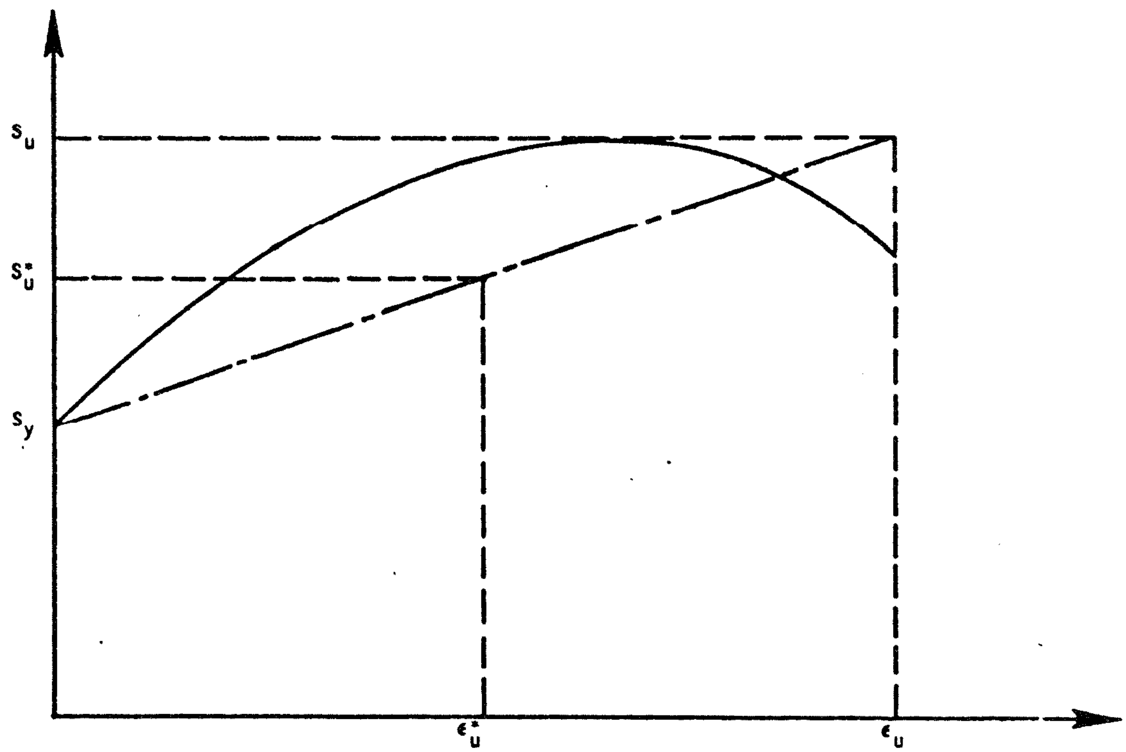
INCREASE OF THE THICKNESS OF A NON-ISOTHERMAL AXIALLY SYMMETRIC JET OF GAS IN A STATIONARY ENVIRONMENT (INFLUENCE OF PARAMETER θ)
FIGURE 3.6-8



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

THE LATERAL BOUNDARY OF
A TWO-PHASE JET

FIGURE 3.6-9



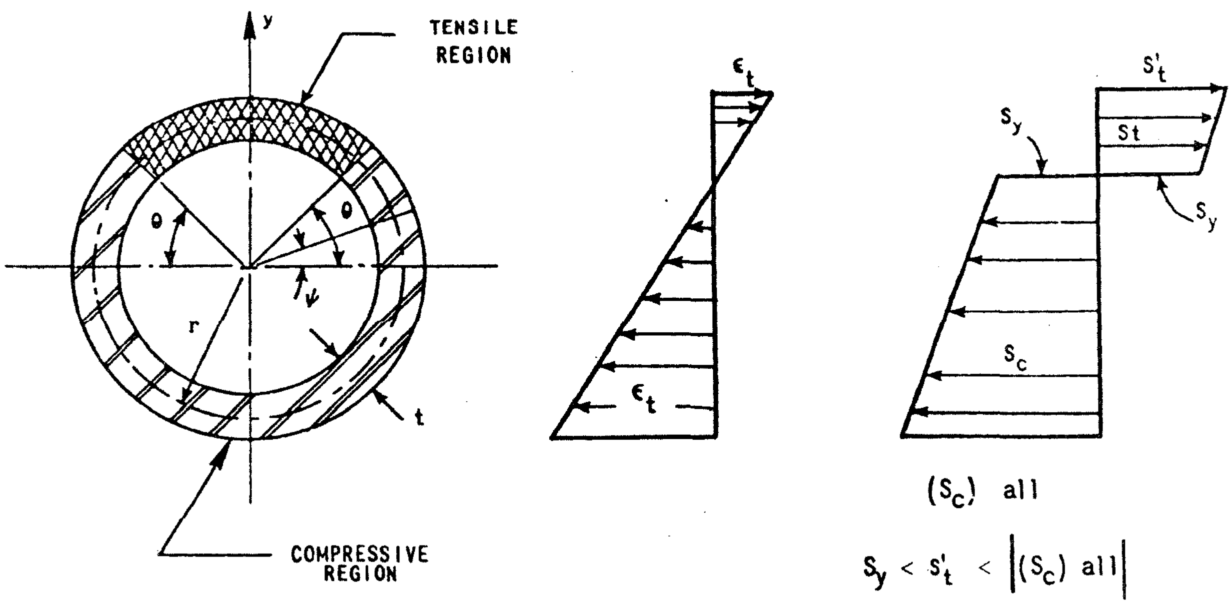
——— Typical Stress-Strain Curve

— - — Simplified Stress-Strain Curve

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

TYPICAL AND SIMPLIFIED
STRESS STRAIN CURVES

FIGURE 3.6-9A



(A) (B) (C)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

TYPICAL STRESS DISTRIBUTION FOR PIPE COLLAPSE

FIGURE 3.6-9B

REFER TO DRAWING

8770-G-819, Sheet 1

Amendment No. 22 (05/07)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR AUXILIARY BUILDING
PIPE RESTRAINTS SH. 1
FIGURE 3.6-10

DELETED

Amendment No. 23 (11/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FIGURE 3.6-11

DELETED

Amendment No. 23 (11/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FIGURE 3.6-12

REFER TO DRAWINGS

8770-G-799, SHEET 5
8770-G-795, SHEET 2

Attachment No. 22 (05/07)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
PIPE RESTRAINTS
FIGURE 3.6-13

REFER TO DRAWINGS

8770-G-799, SHEETS 5 & 15

Amendment No. 22 (05/07)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
PIPE RESTRAINTS
FIGURE 3.6-14

REFER TO DRAWINGS

8770-G-799, SHEETS 7 & 14

Amendment No. 22 (05/07)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
PIPE RESTRAINTS
FIGURE 3.6-15

REFER TO DRAWINGS

8770-G-799, SHEETS 5 & 15
8770-G-795, SHEET 2

Amendment No. 22 (05/07)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
PIPE RESTRAINTS
FIGURE 3.6-16

REFER TO DRAWINGS

8770-G-799, SHEETS 5, 13 & 15
8770-G-795, SHEET 2

Amendment No. 22 (05/07)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
PIPE RESTRAINTS
FIGURE 3.6-17

REFER TO DRAWING

8770-G-799, SHEET 8

Amendment No. 22 (05/07)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
PIPE RESTRAINTS
FIGURE 3.6-18

FIGURES 3.6-19 THROUGH 3.6-28

HAVE BEEN DELETED

Amendment No. 23 (11/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FIGURE 3.6-19

REFER TO DRAWINGS

8770-G-819, SHEETS 1& 2

Amendment No. 18, (04/01)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR AUXILIARY BUILDING
SAFETY INJECTION SYSTEM
PIPING P.W. RESTR'S
FIGURE 3.6-29

DELETED

Amendment No. 23 (11/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FIGURE 3.6-30

DELETED

Amendment No. 23 (11/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

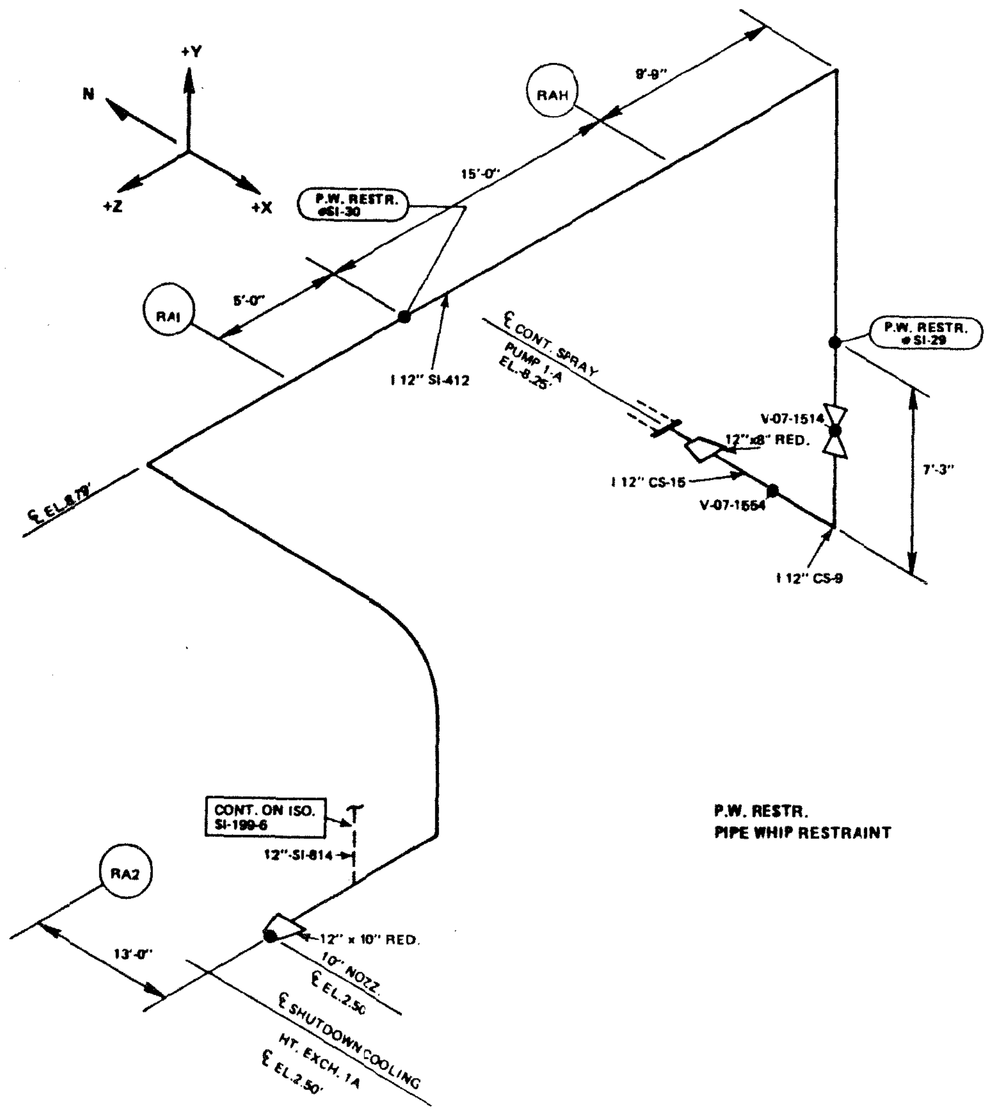
FIGURE 3.6-31

DELETED

Amendment No. 23 (11/08)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

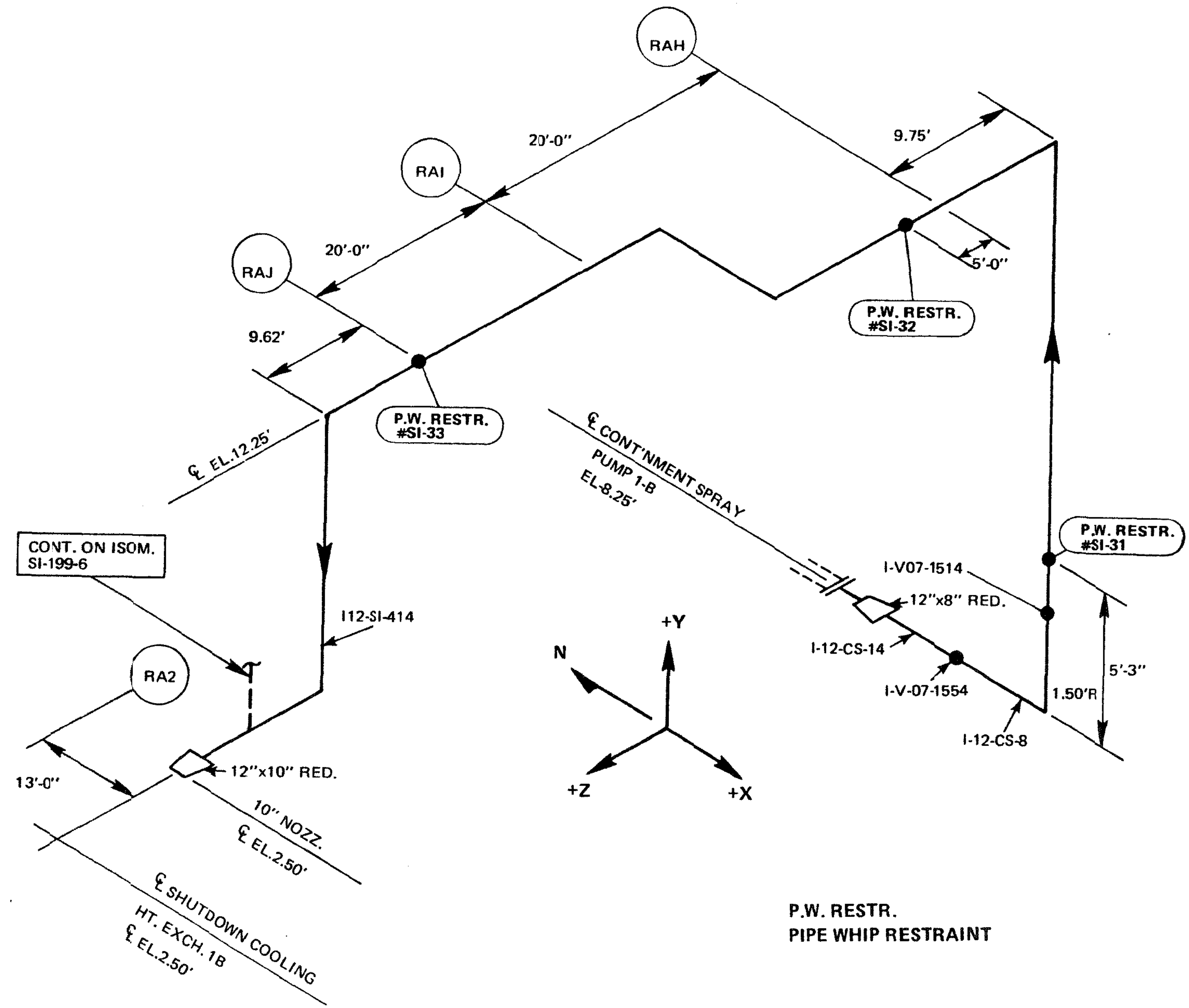
FIGURE 3.6-32



P.W. RESTR.
PIPE WHIP RESTRAINT

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR AUX. BUILDING
CONTAINMENT SPRAY
PUMPS - DISCH. PIPING P.W. RESTR'S
FIGURE 3.6-33

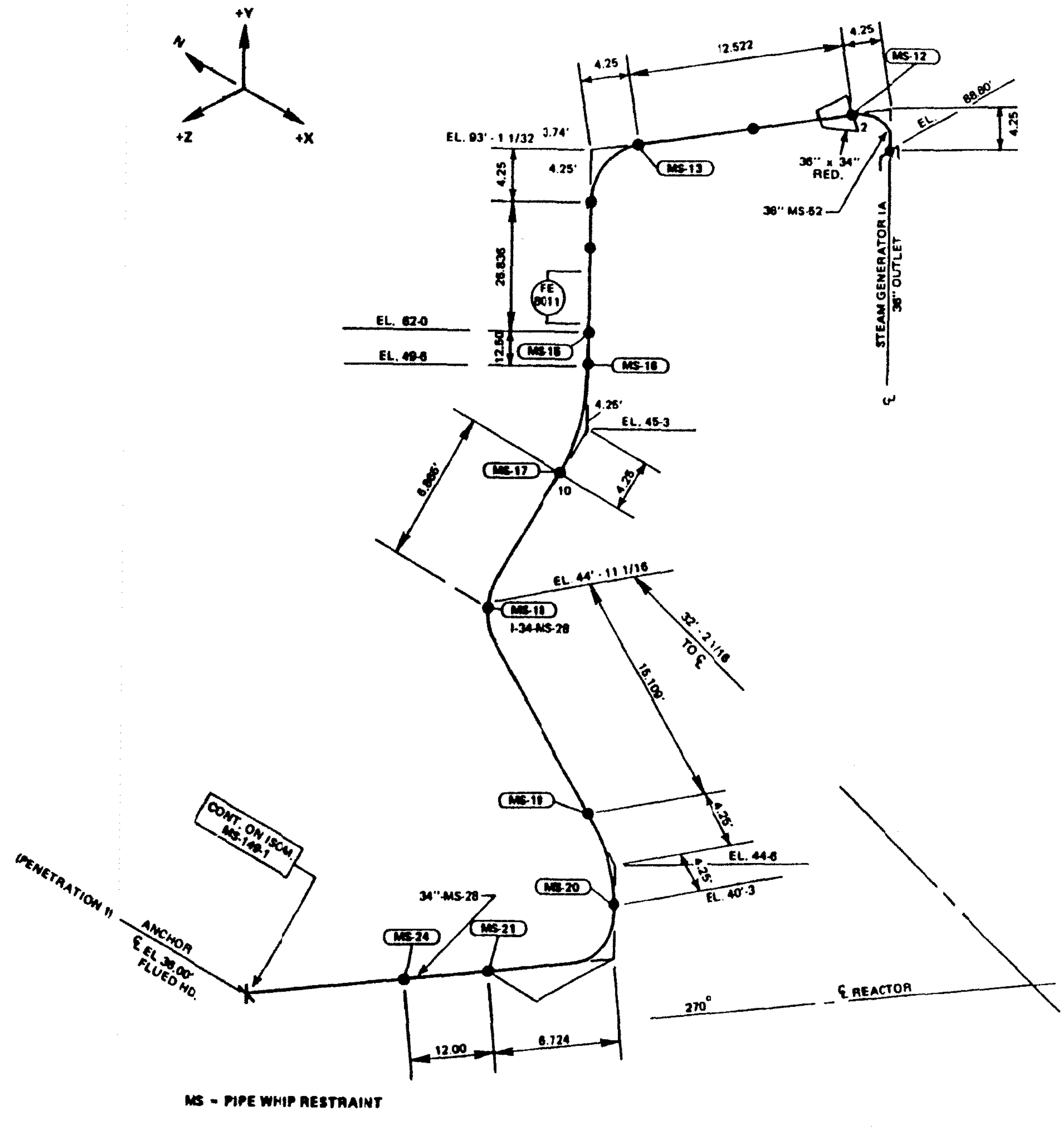


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

REACTOR AUX. BUILDING
CONTAINMENT SPRAY PUMPS
DISCH. PIPING P.W. RESTR'S

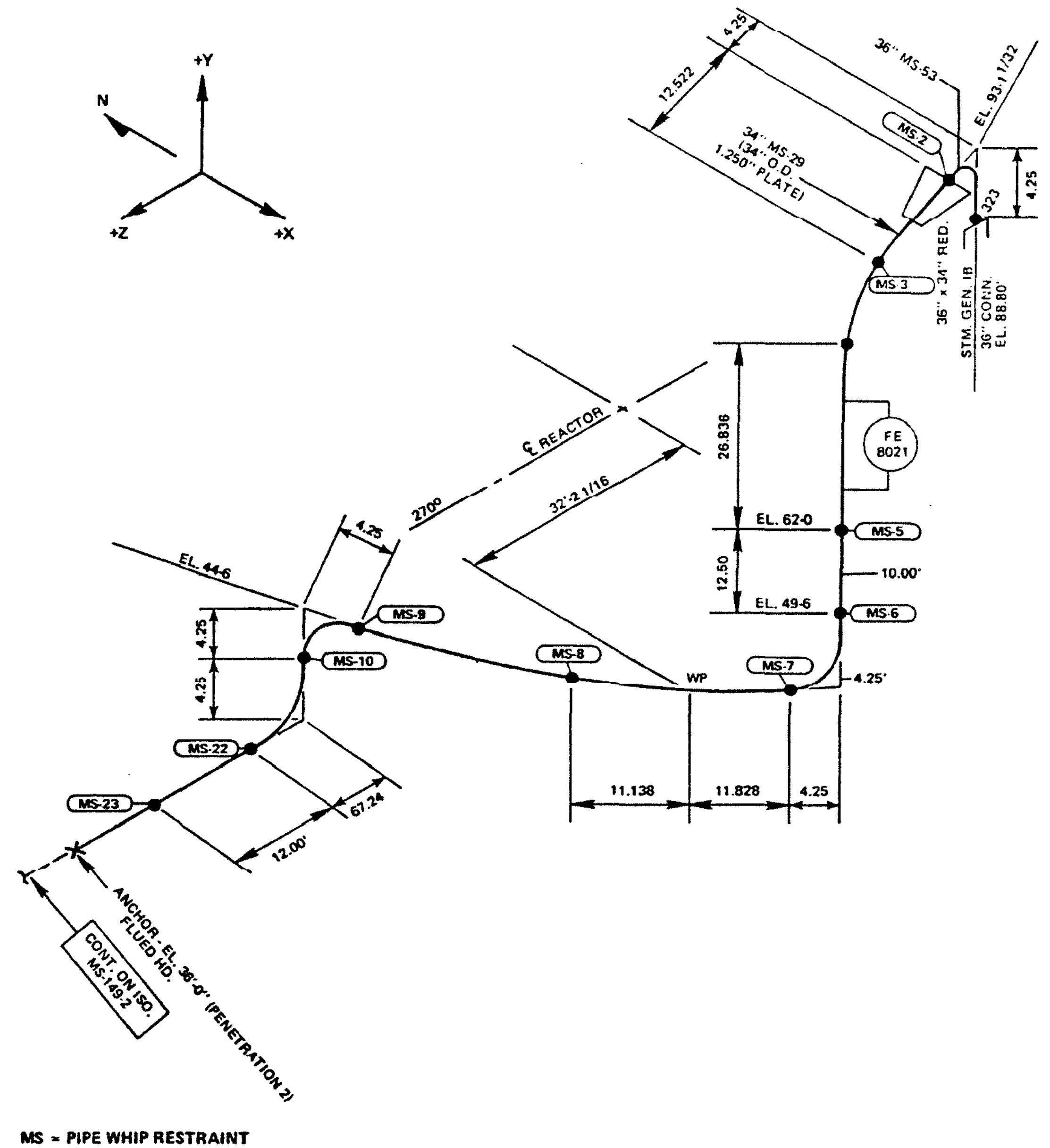
FIGURE 3.6-34



Amendment No. 16, (1/98)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
MAIN STEAM PIPING RESTRAINTS
FIGURE 3.6-35

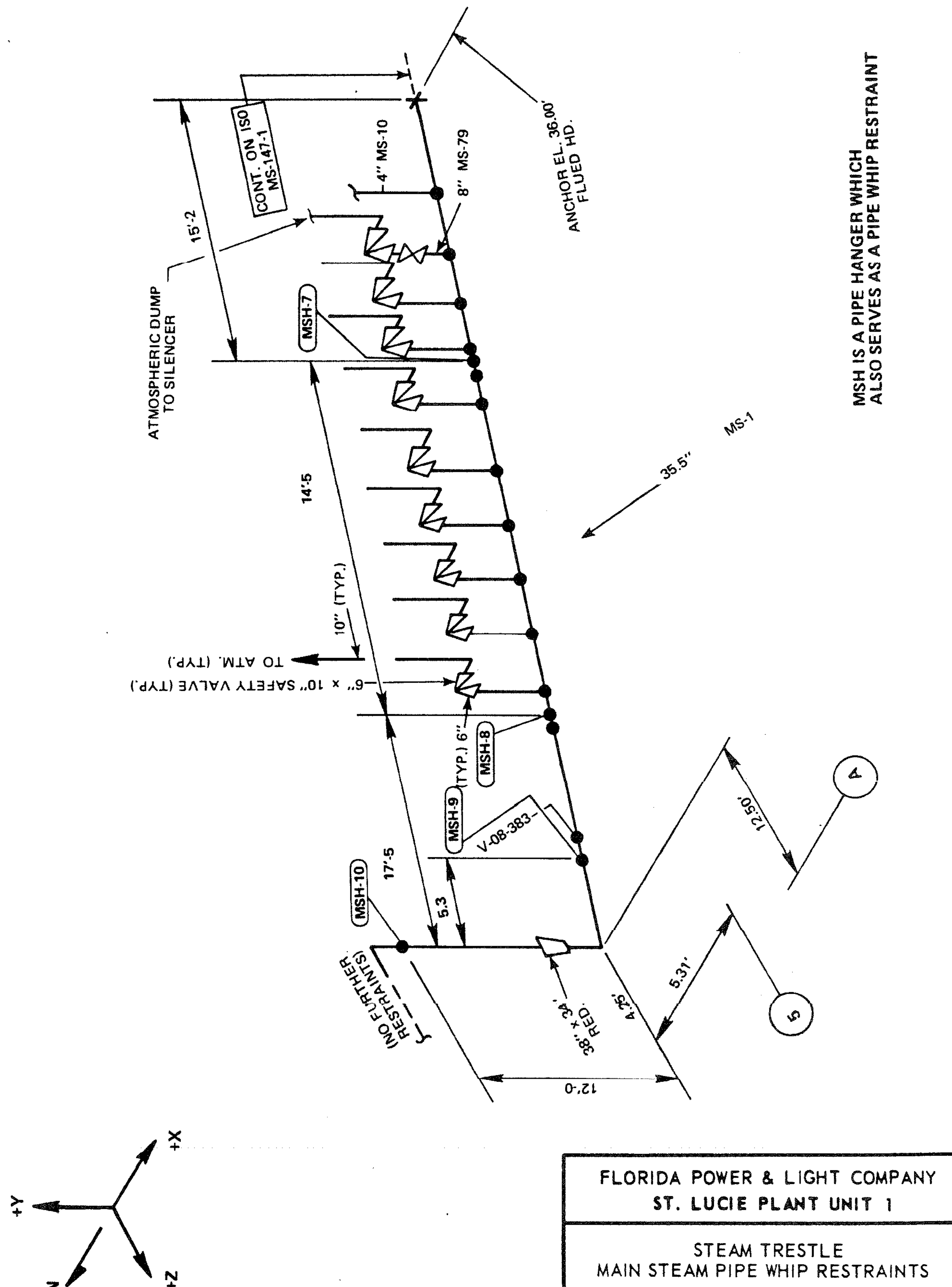


Amendment No. 16, (1/98)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

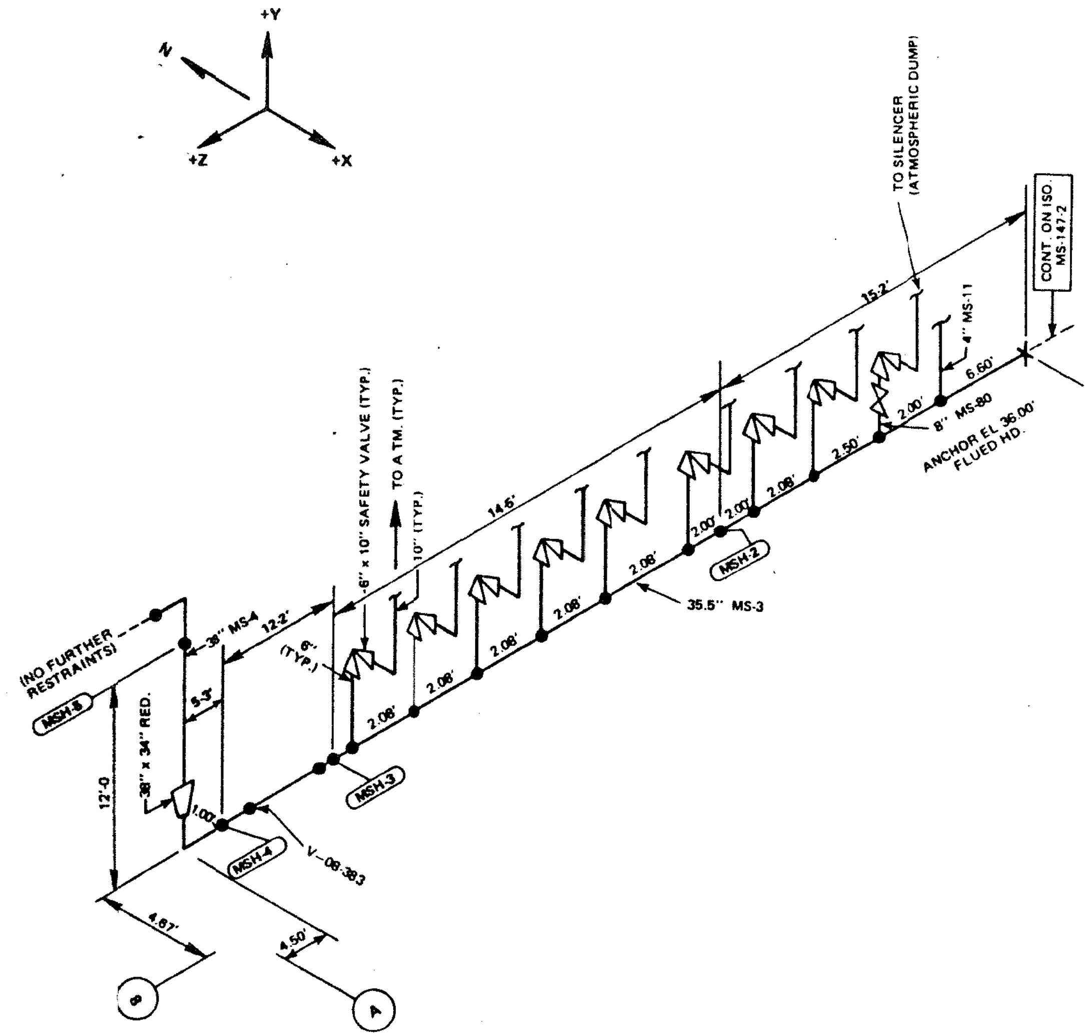
REACTOR BUILDING MAIN STEAM PIPE WHIP RESTRAINTS

FIGURE 3.6-36



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

STEAM TRESTLE
MAIN STEAM PIPE WHIP RESTRAINTS
FIGURE 3.6-37

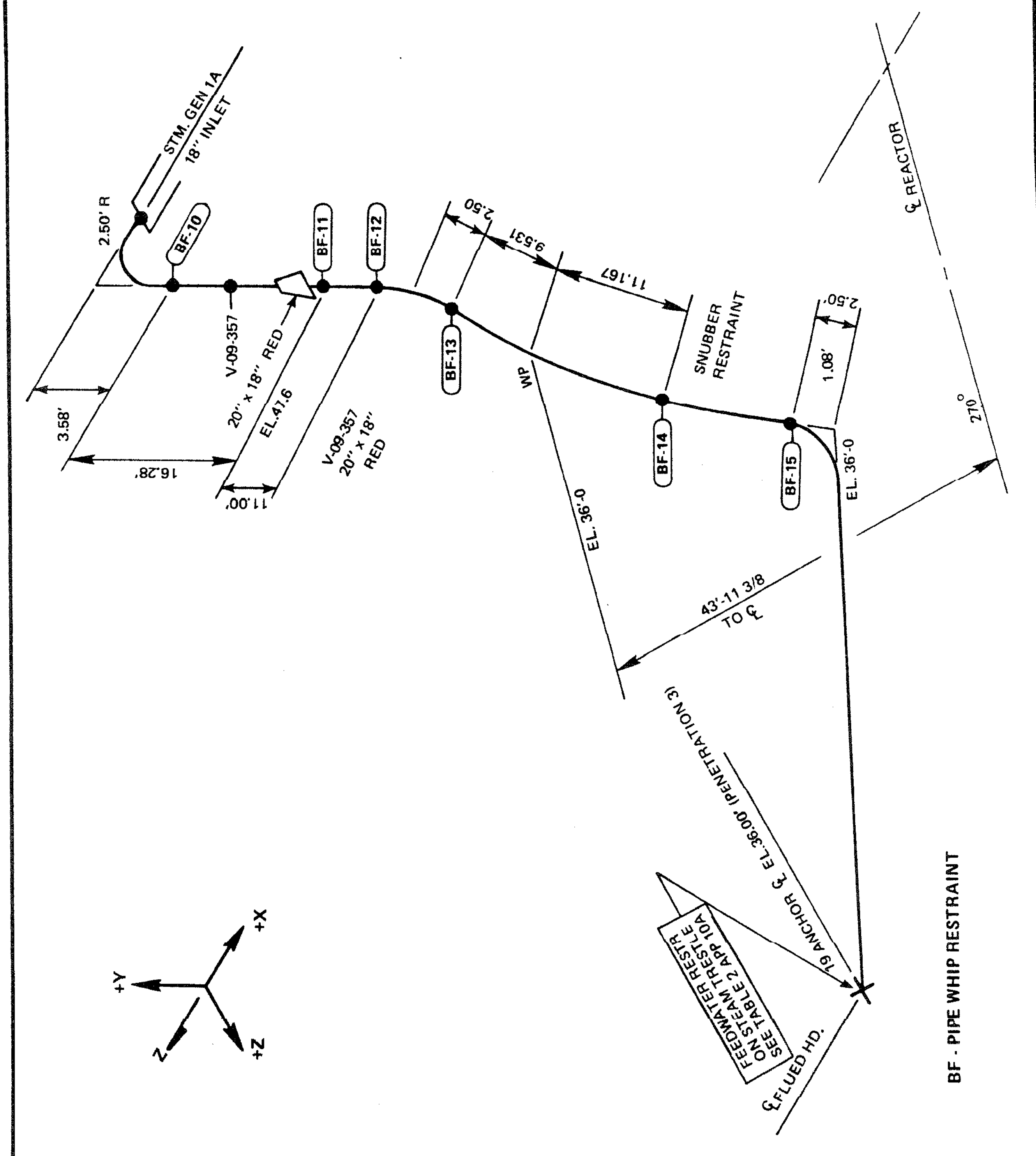


MSH IS A PIPE HANGER WHICH ALSO
SERVES AS A PIPE WHIP RESTRAINT

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

STEAM TRESTLE
MAIN STEAM PIPE WHIP RESTRAINTS

FIGURE 3.6-38

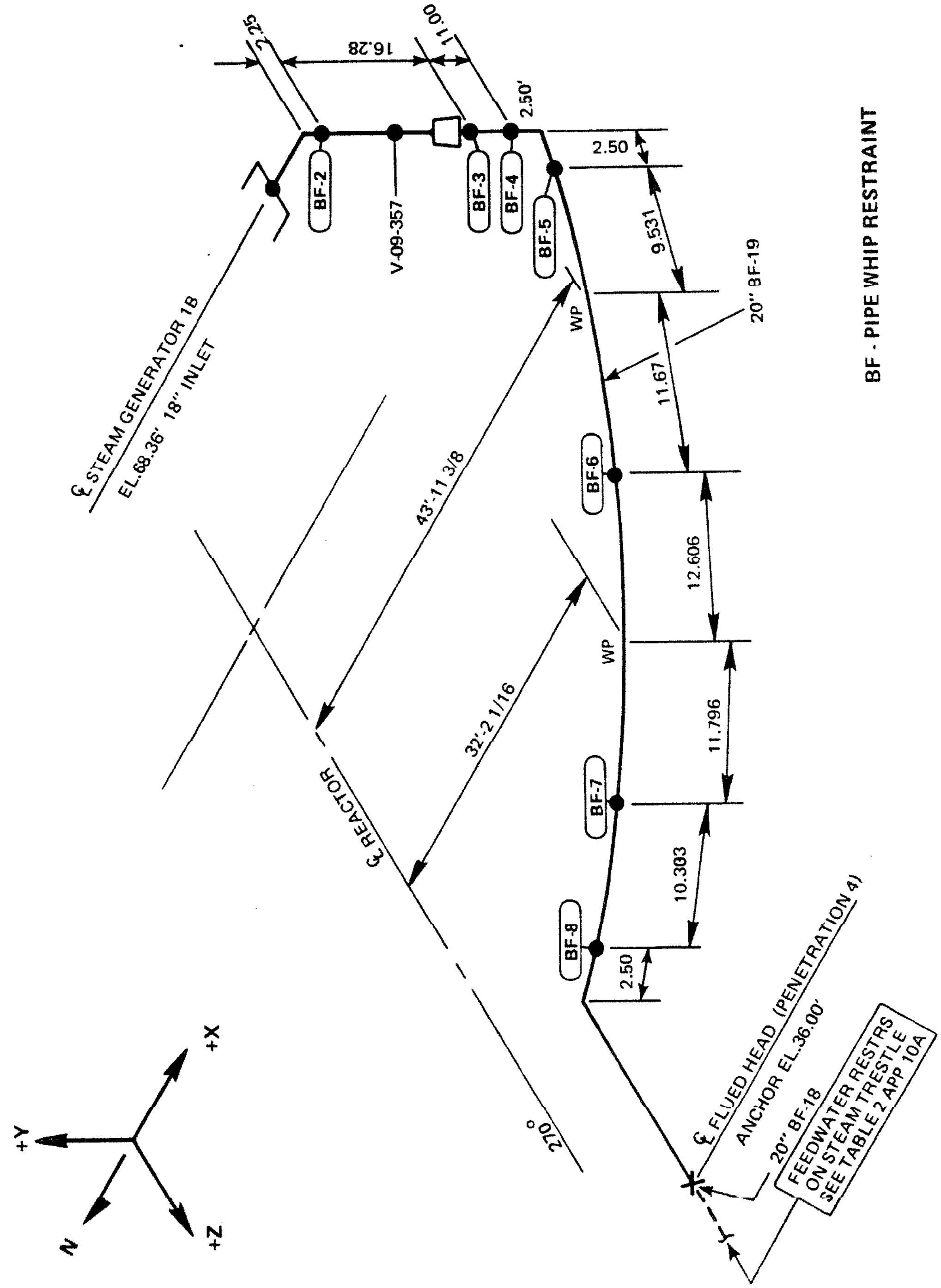


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

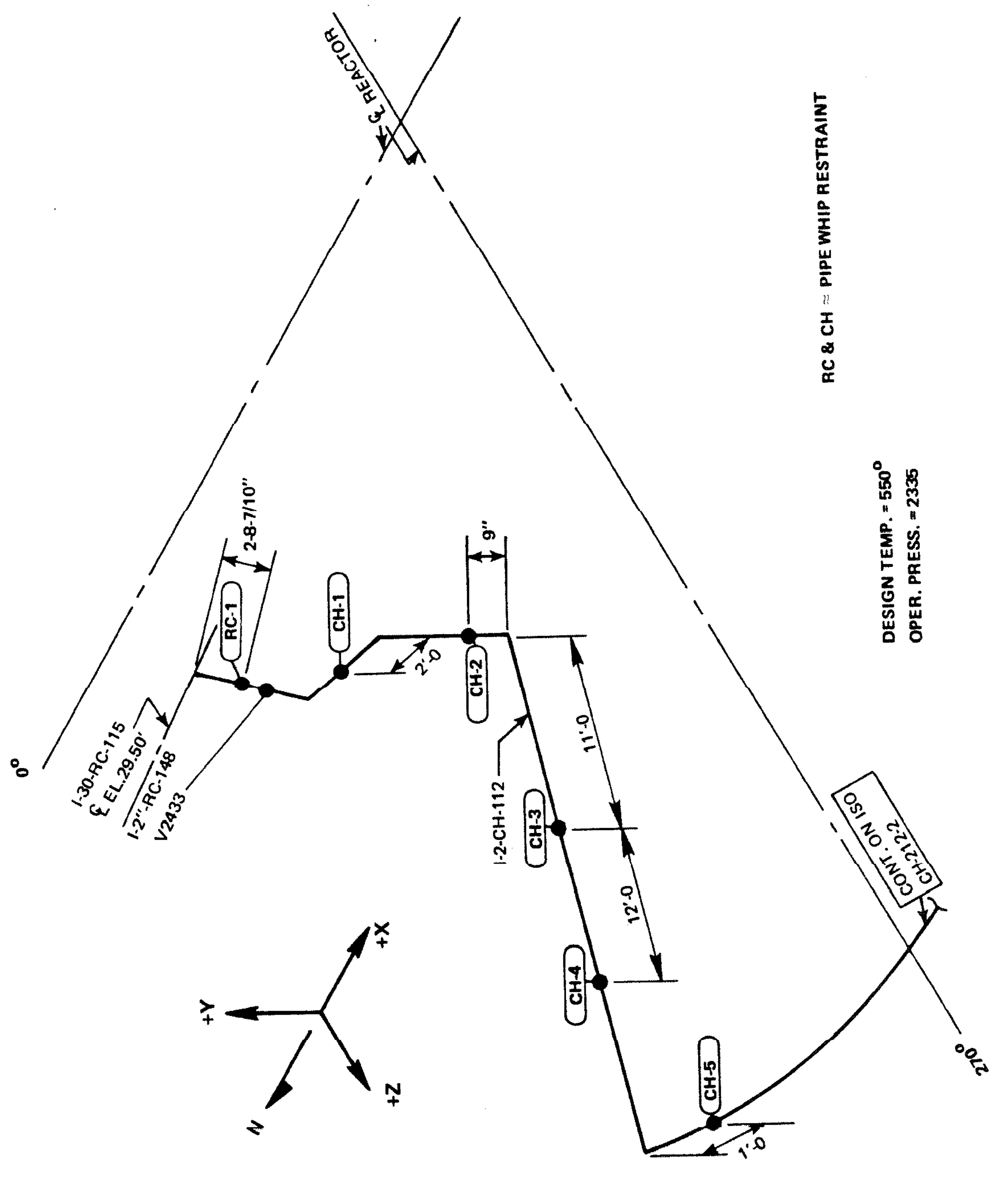
REACTOR BUILDING
FEEDWATER PIPE WHIP RESTRAINTS

FIGURE 3.6-39



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

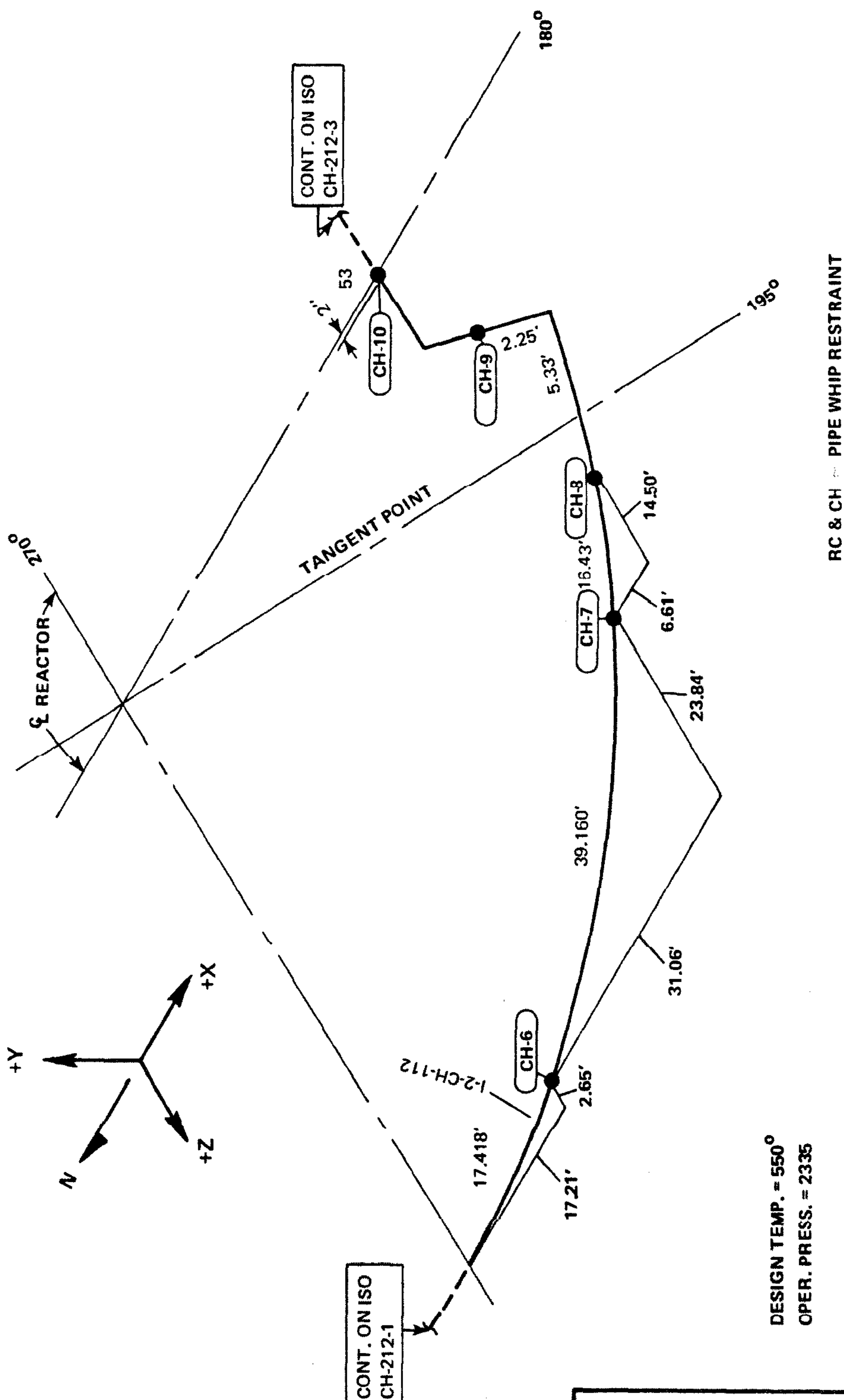
REACTOR BUILDING
FEEDWATER PIPING
PIPE WHIP RESTRAINTS
FIGURE 3.6-40



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
CHEMICAL VOLUME CONTROL P.W. RESTR'S

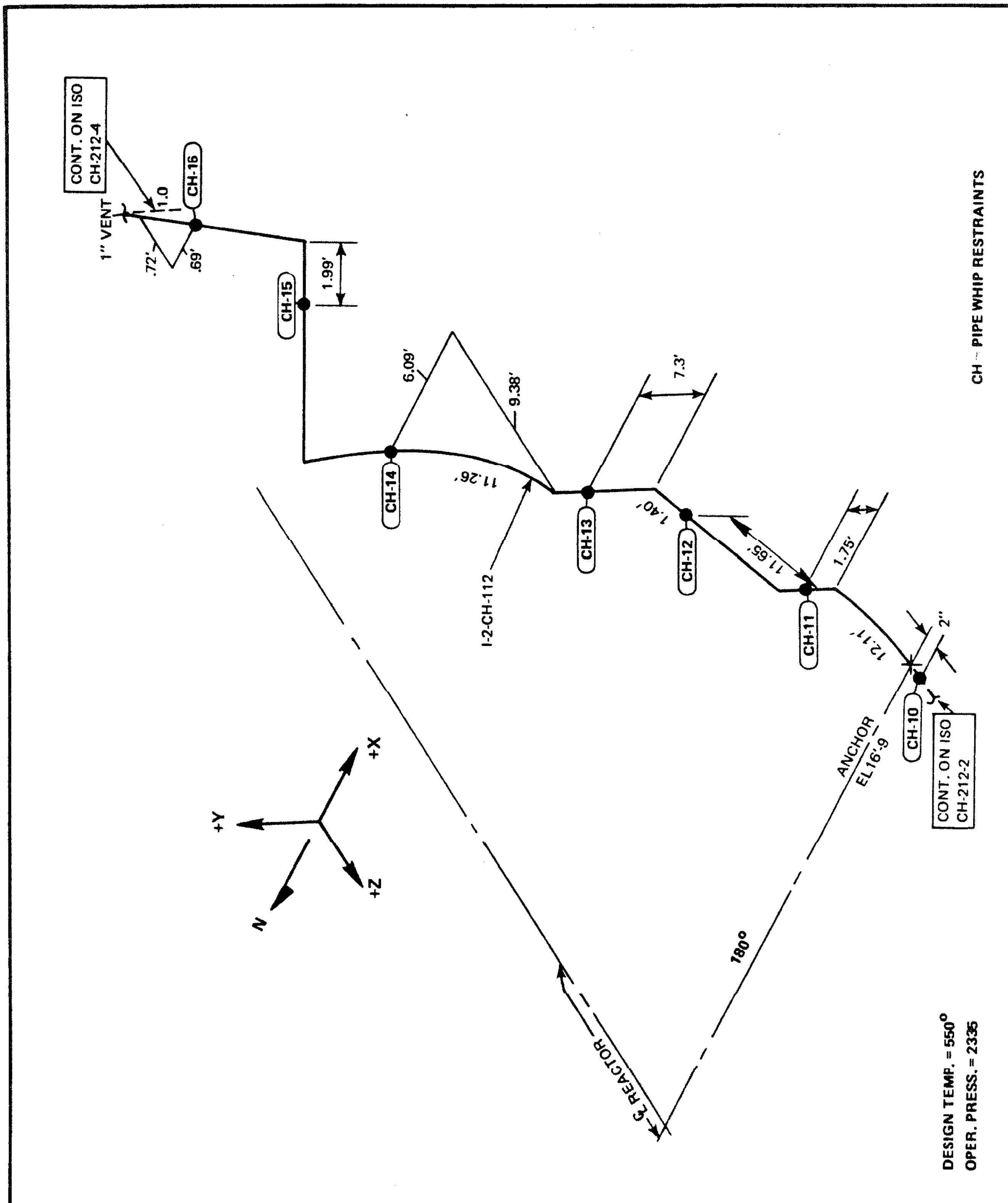
FIGURE 3.6-41



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

CVCS PIPE WHIP RESTRAINTS

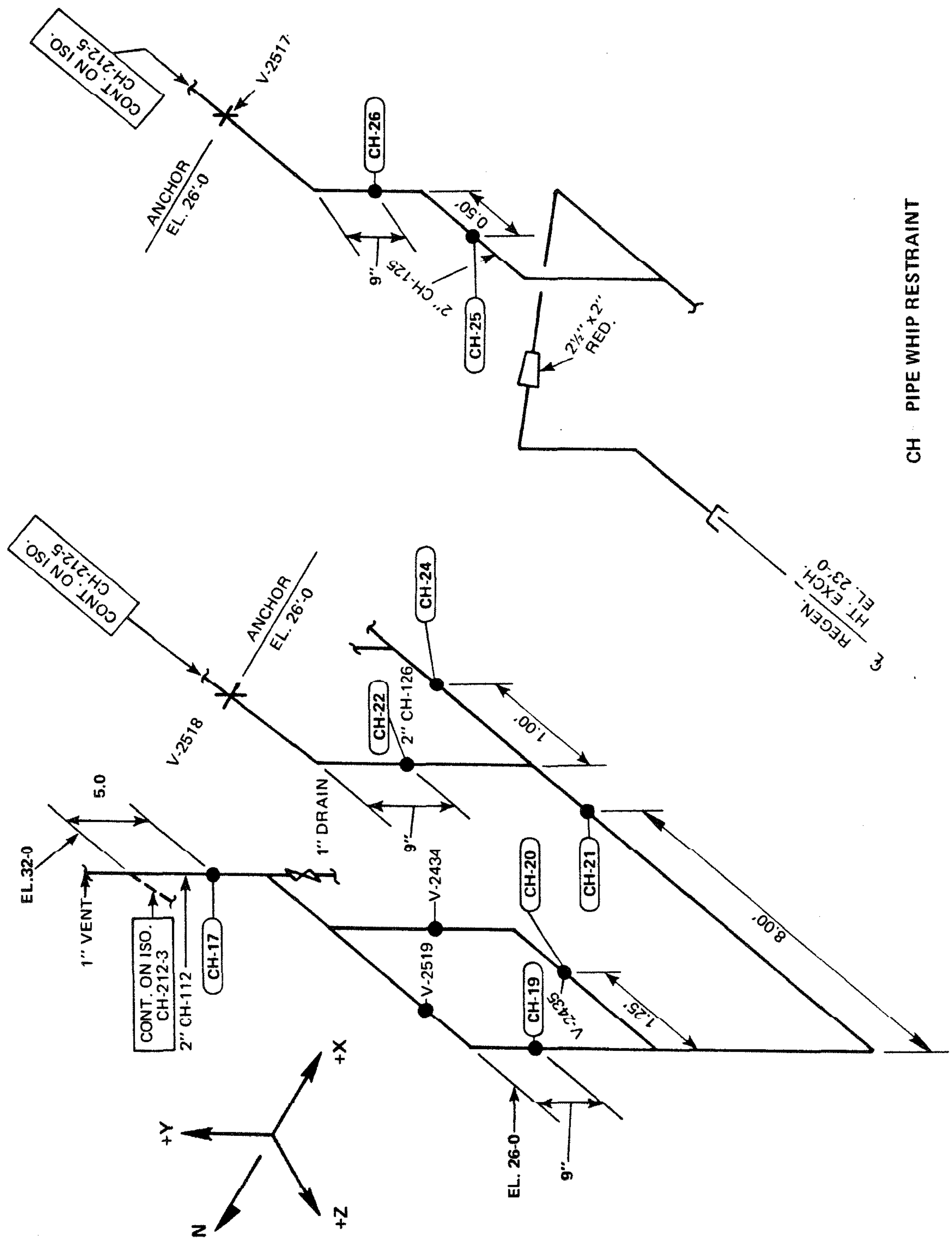
FIGURE 3.6-42



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
CVCS PIPE WHIP RESTRAINTS

FIGURE 3.6-43

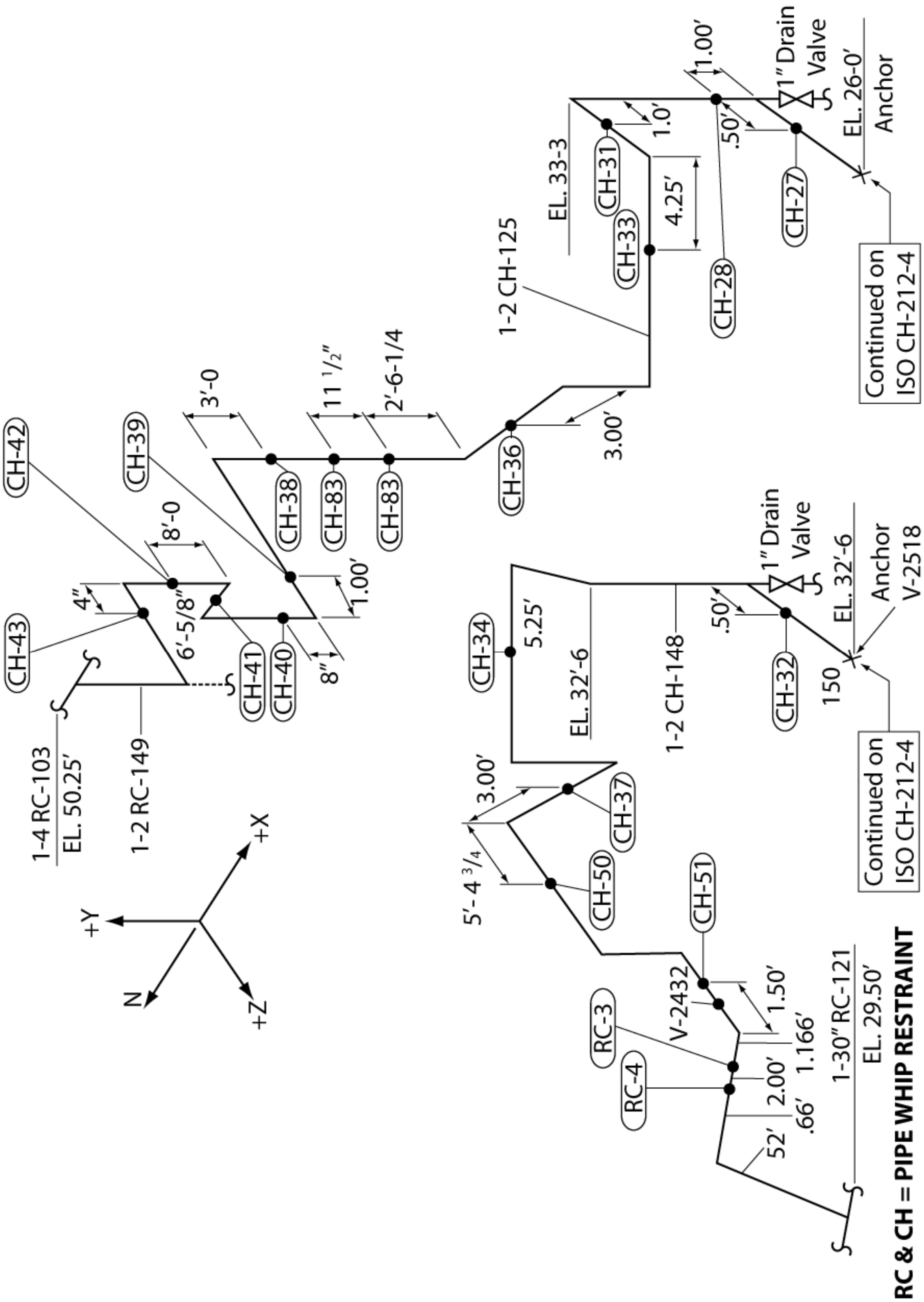


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
CVCS PIPE WHIP RESTRAINTS

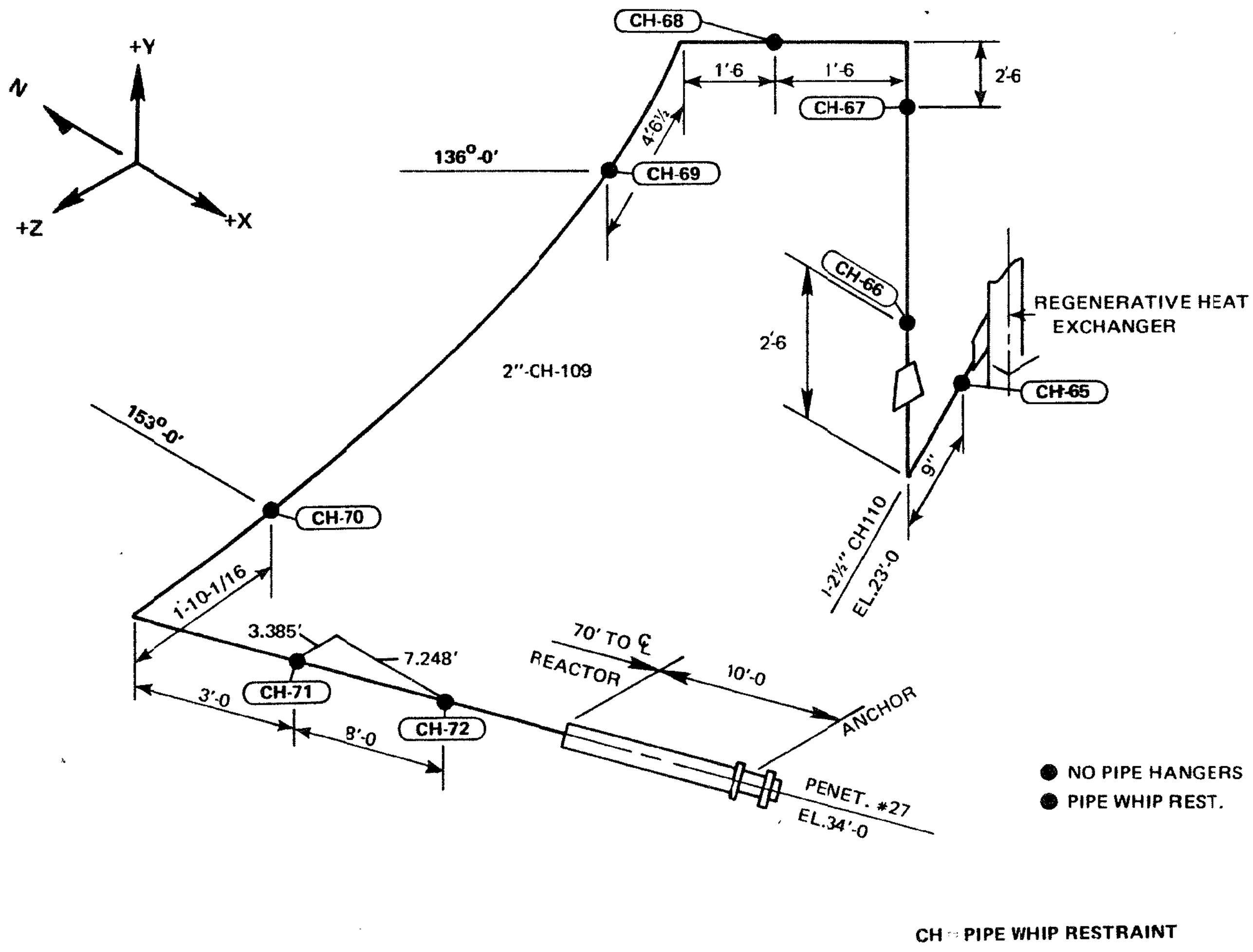
FIGURE 3.6-44



Amendment No. 24 (06/10)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

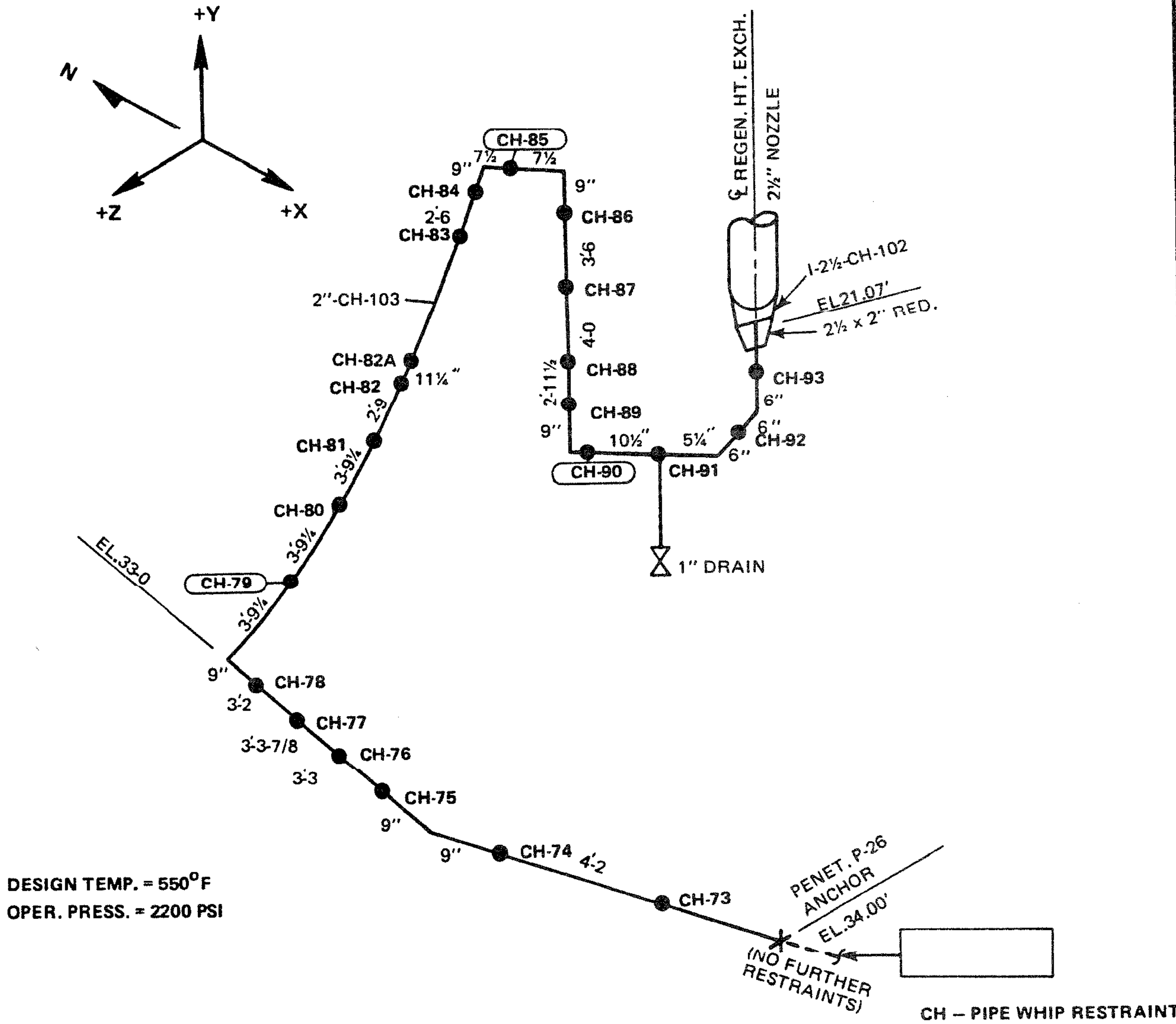
REACTOR BUILDING
CVCS PIPE WHIP RESTRAINTS
FIGURE 3.6-45



FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

REACTOR BUILDING
CVCS PIPE WHIP RESTRAINTS
FIGURE 3.6-46



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING

CVCS PIPE WHIP RESTRAINTS

FIGURE 3.6-47

REFER TO DRAWING

8770-G-799, Sheets 5 & 8

Amendment No. 16, (1/98)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING PIPE RESTRAINTS
(SG 1A BLOWDOWN PIPING)

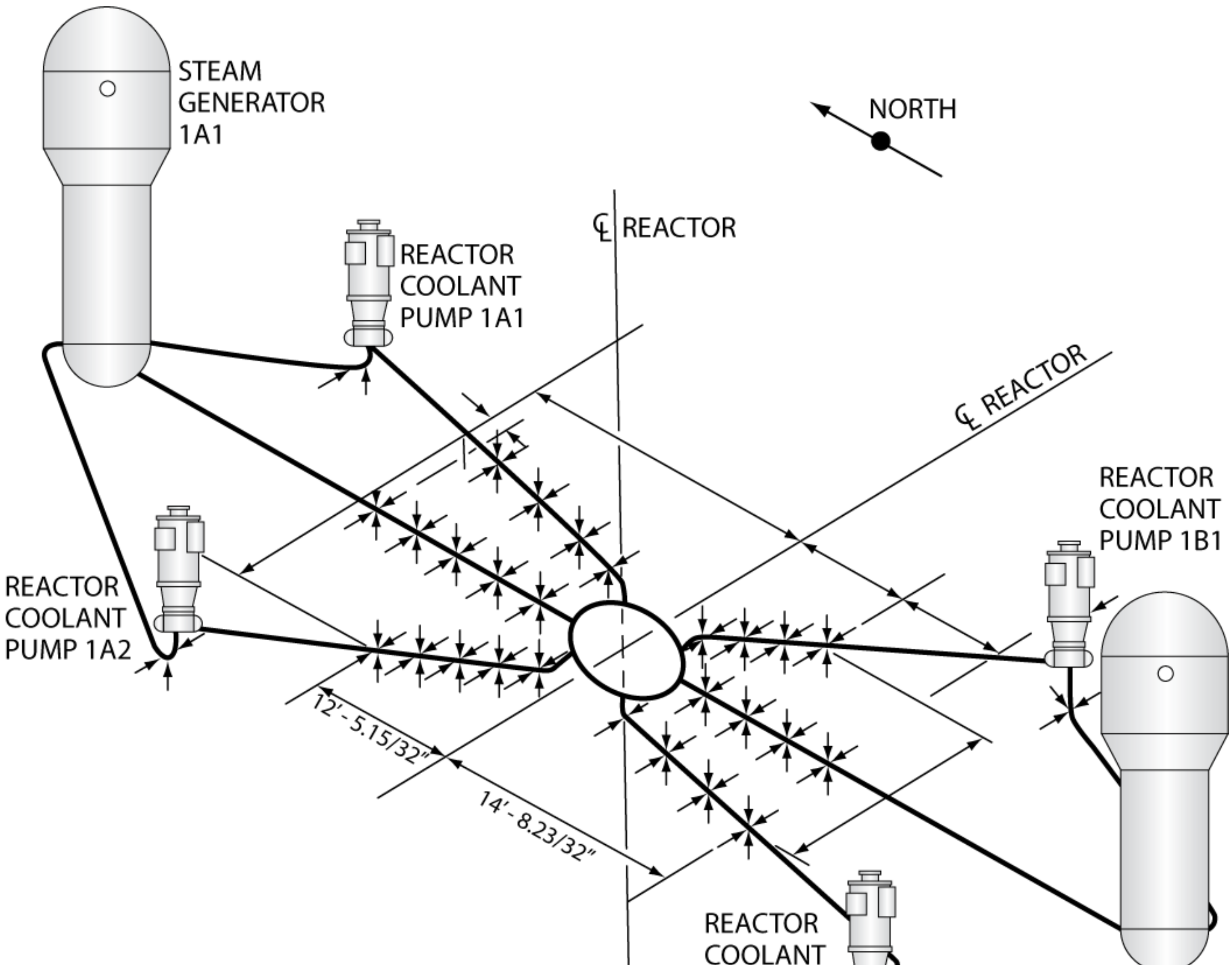
FIGURE 3.6-48

REFER TO DRAWING

8770-G-799, Sheets 10, 11 & 15

Amendment No. 16, (1/98)

FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT UNIT 1**
REACTOR BUILDING PIPE RESTRAINTS (SG 1B BLOWDOWN PIPING) **FIGURE 3.6-49**

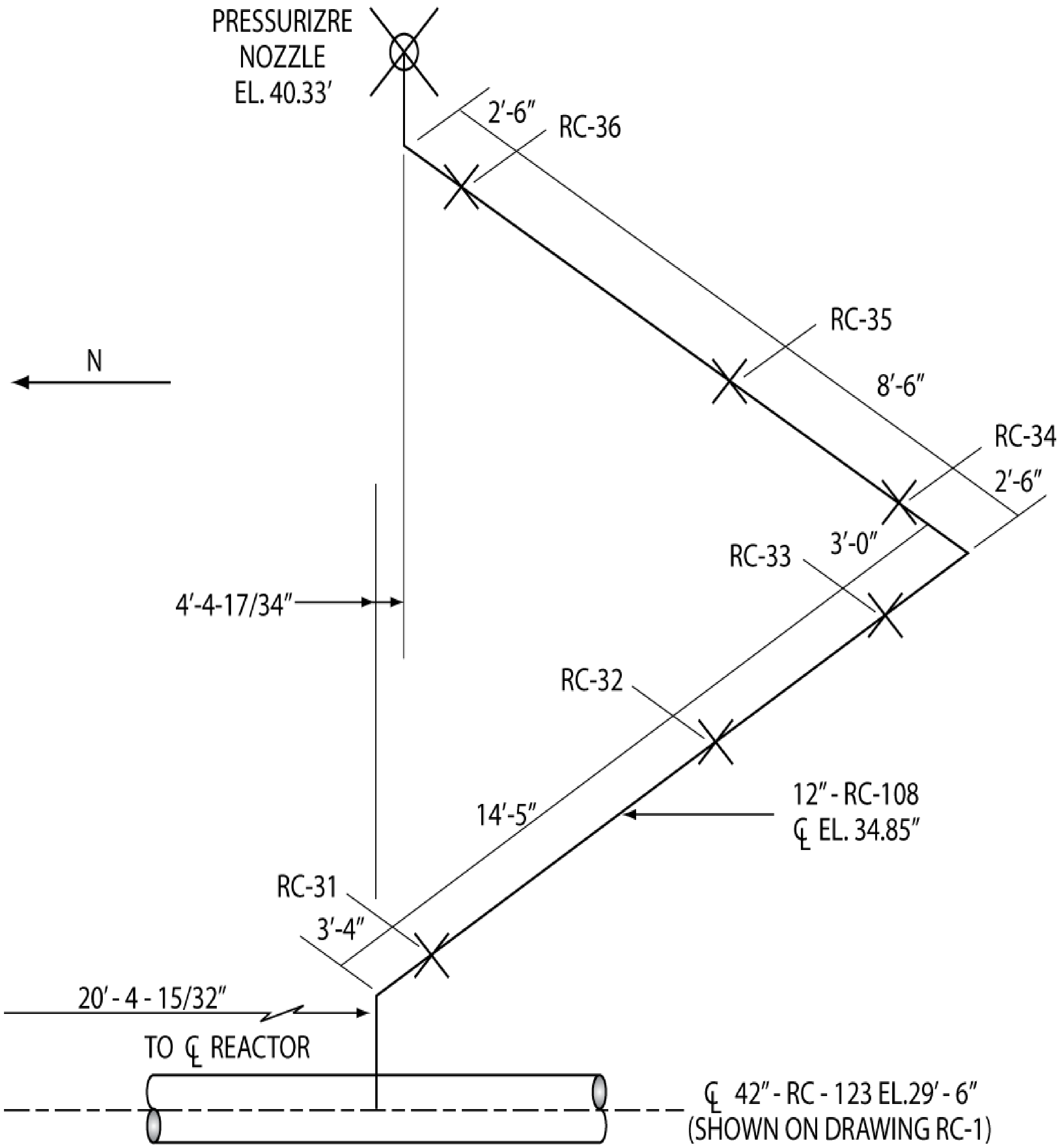


Information provided in this drawing is historical. Refer to page 3.6-1 of this USFAR for the status of RC Loop pipe whip restraints.

Amendment No. 26 (11/13)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

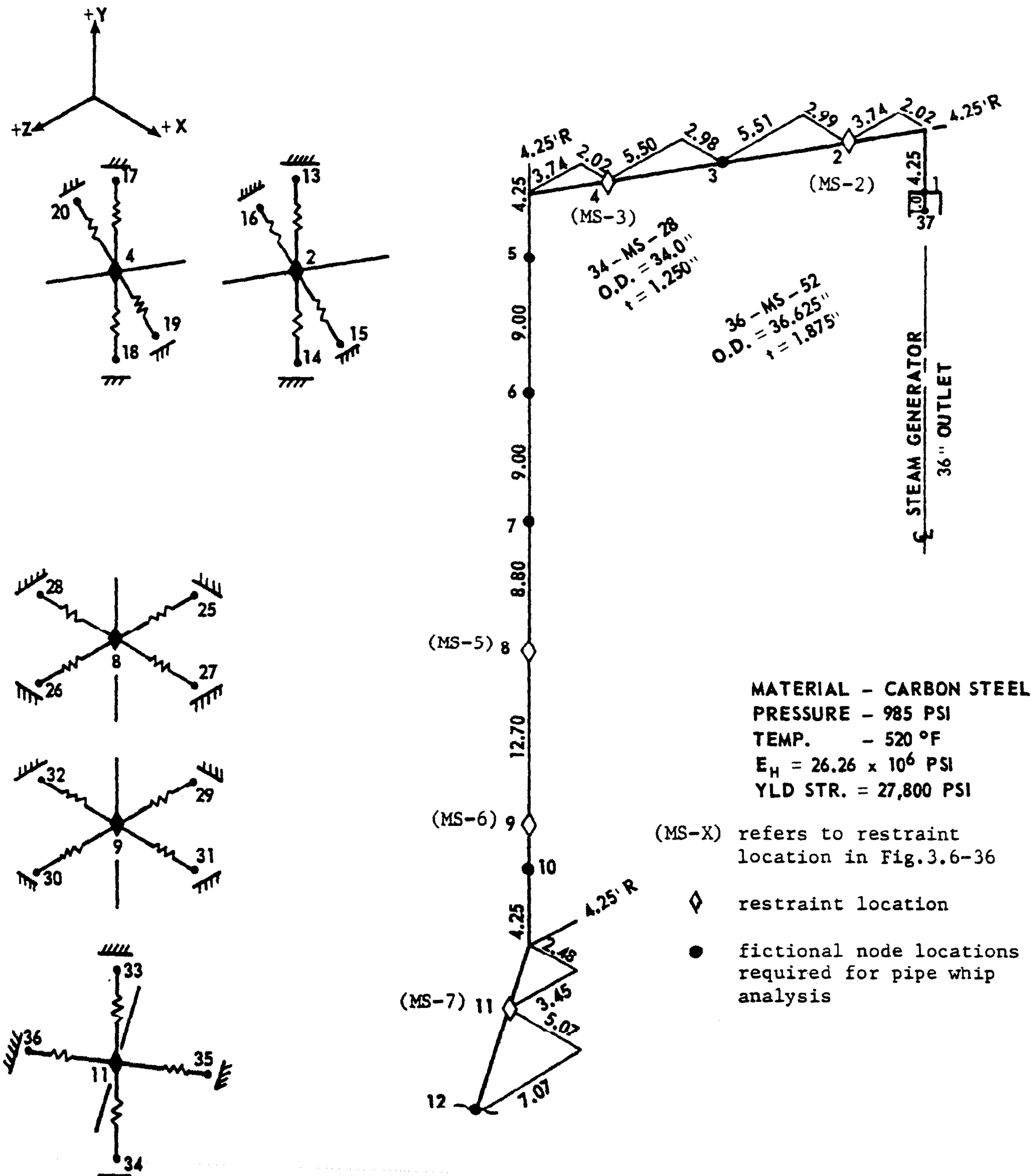
REACTOR COOLANT LOOP
PIPE WHIP RESTRAINTS
(Historical)
FIGURE 3.6-50



Amendment No. 24 (06/10)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

PRESSUR SURGE LINE
PIPE WHIP RESTRAINTS
FIGURE 3.6-51



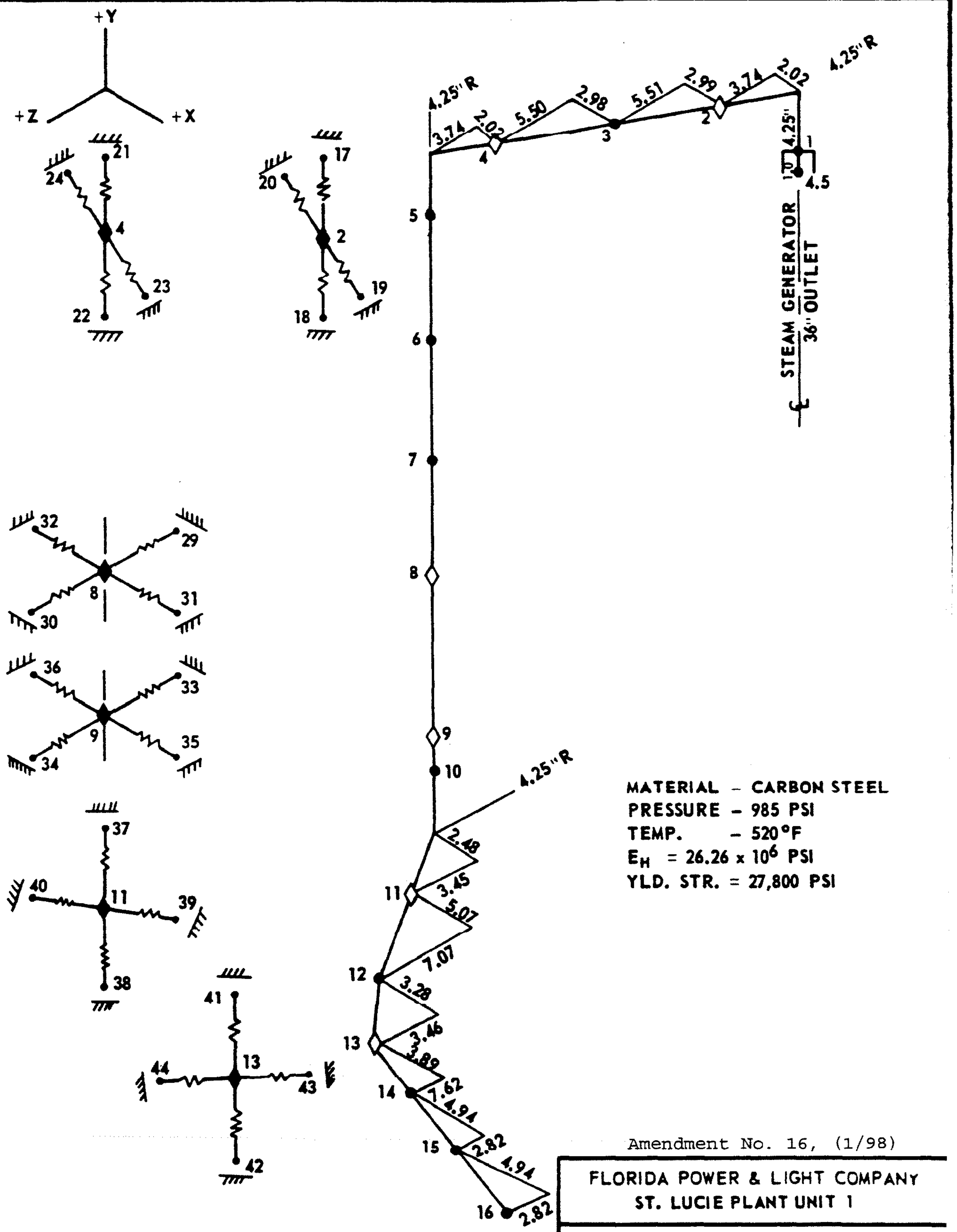
REF. ISOMETRIC: MS - 147-1

Amendment No. 16, (1/98)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAIN STEAM LINE INSIDE CONTAINMENT
GUILLOTINE BREAK AT NODE # 12

FIGURE 3.6-52



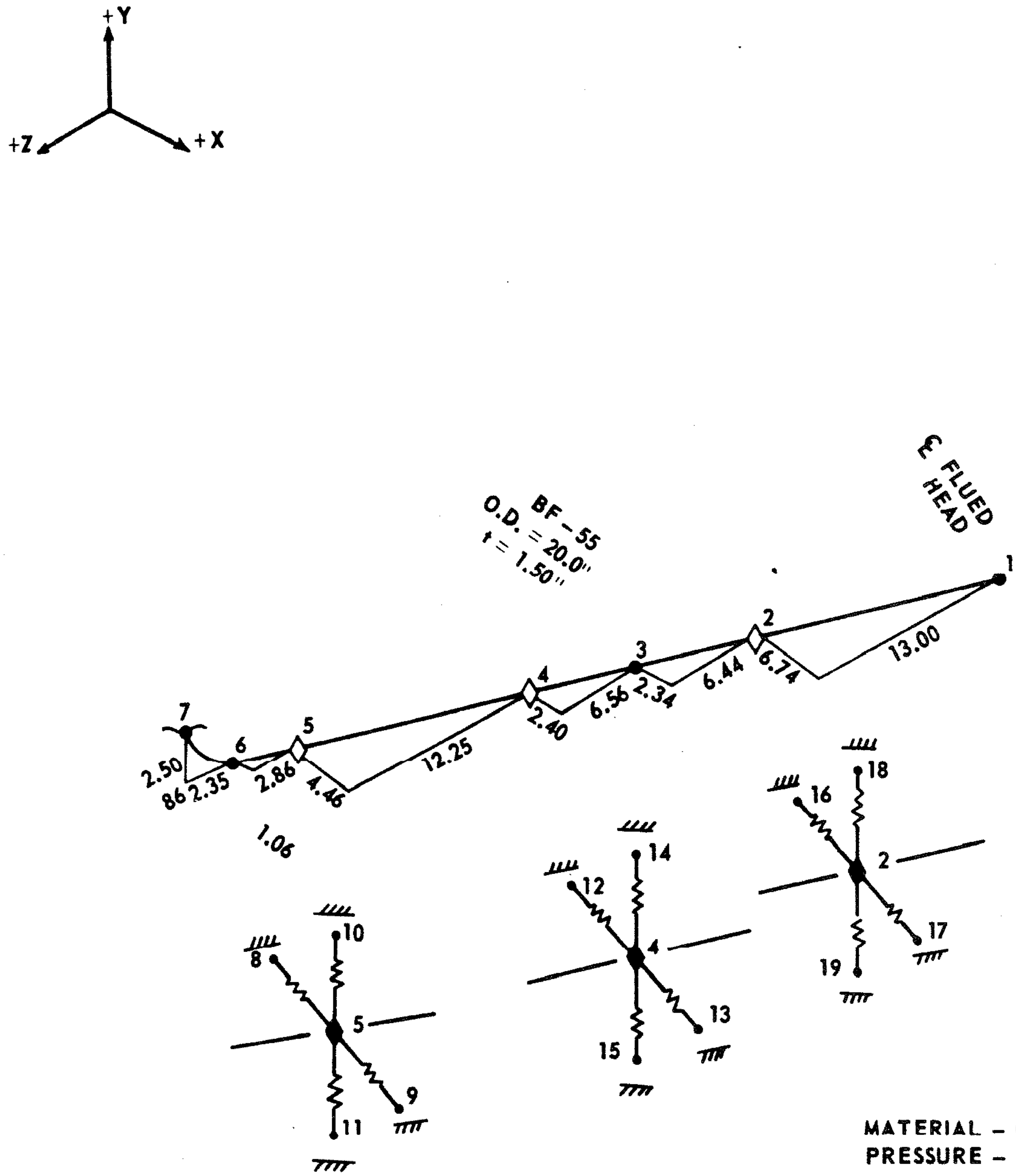
REF. ISOMETRIC: MS - 147 - 1

Amendment No. 16, (1/98)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAIN STEAM LINE INSIDE CONTAINMENT
GUILLOTINE BREAK AT NODE #16

FIGURE 3.6-53



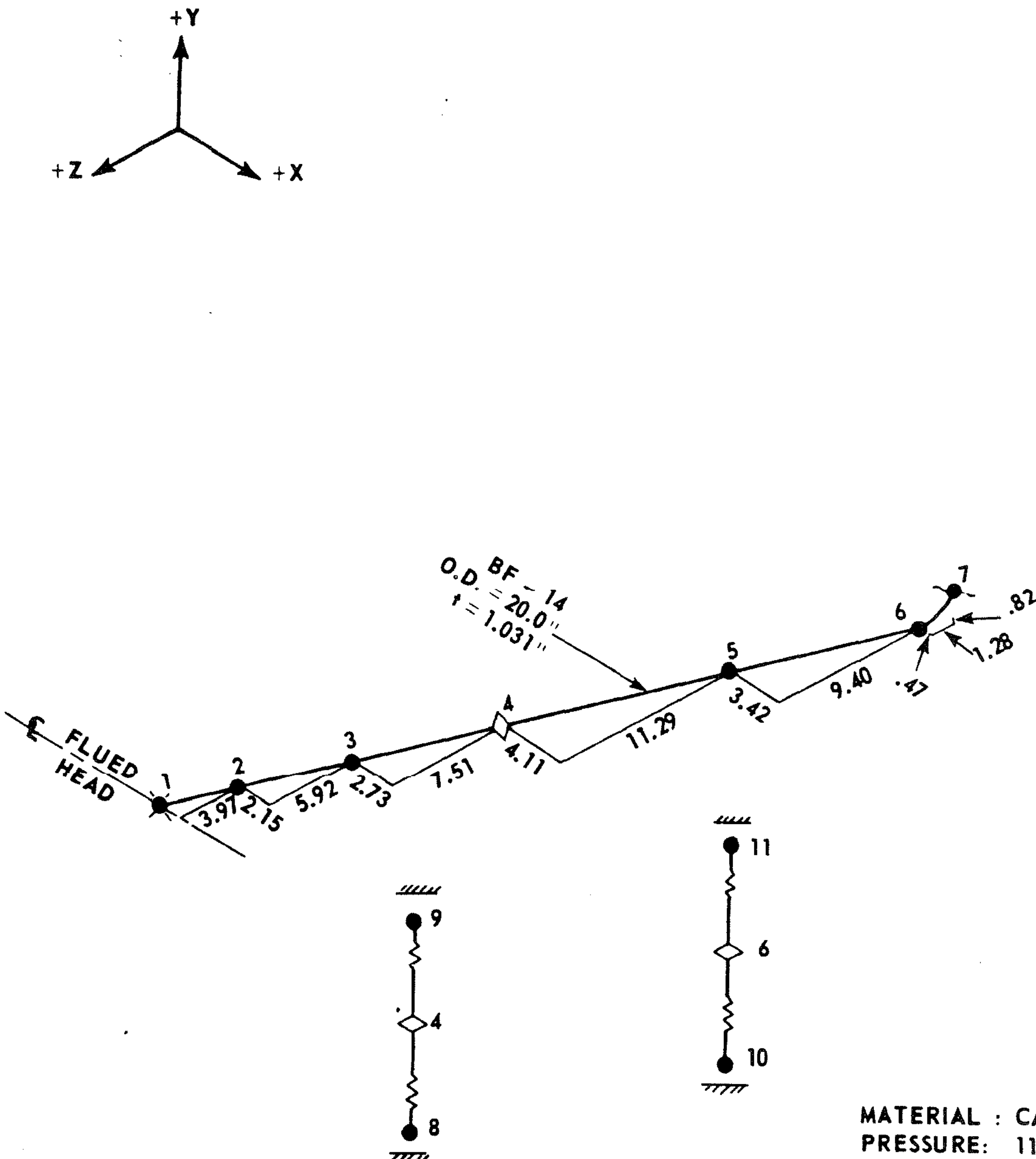
MATERIAL – CARBON STEEL
PRESSURE – 1100 PSI
TEMP. – 440°F
E_H = 26.76 x 10^6 PSI
YLD. STR. = 29,320 PSI

REF. ISOMETRIC: BF – 149 – 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

BOILER FEEDWATER LINE OUTSIDE CONTAINMENT GUILLOTINE BREAK AT NODE #7

FIGURE 3.6-54



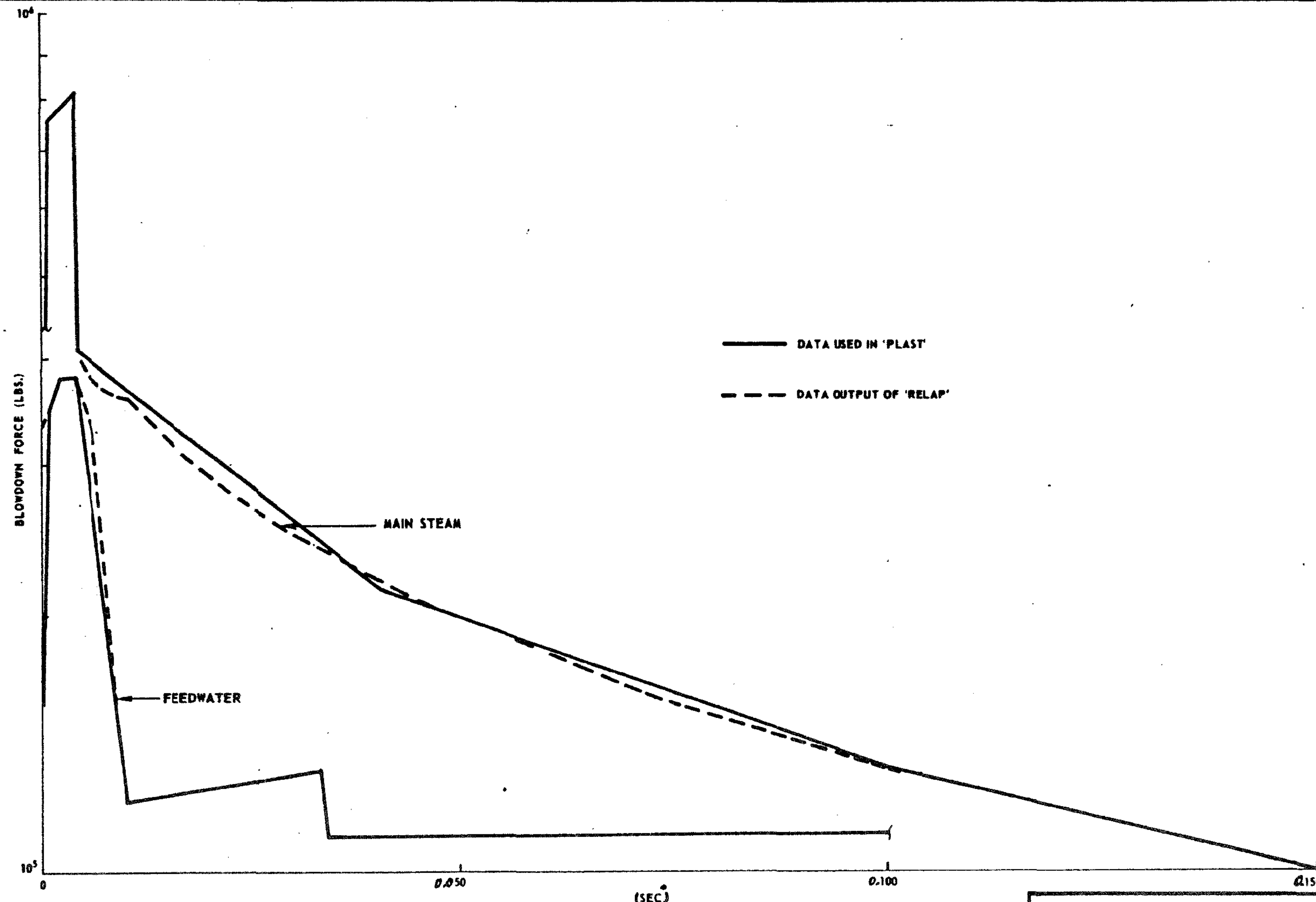
MATERIAL : CARBON STL.
PRESSURE: 1100 PSI
TEMP: 440 °F
E_H = 26.76×10^6 PSI
YLD. STR. = 29,320 PSI

REF. ISOMETRIC: BF - 147 - 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

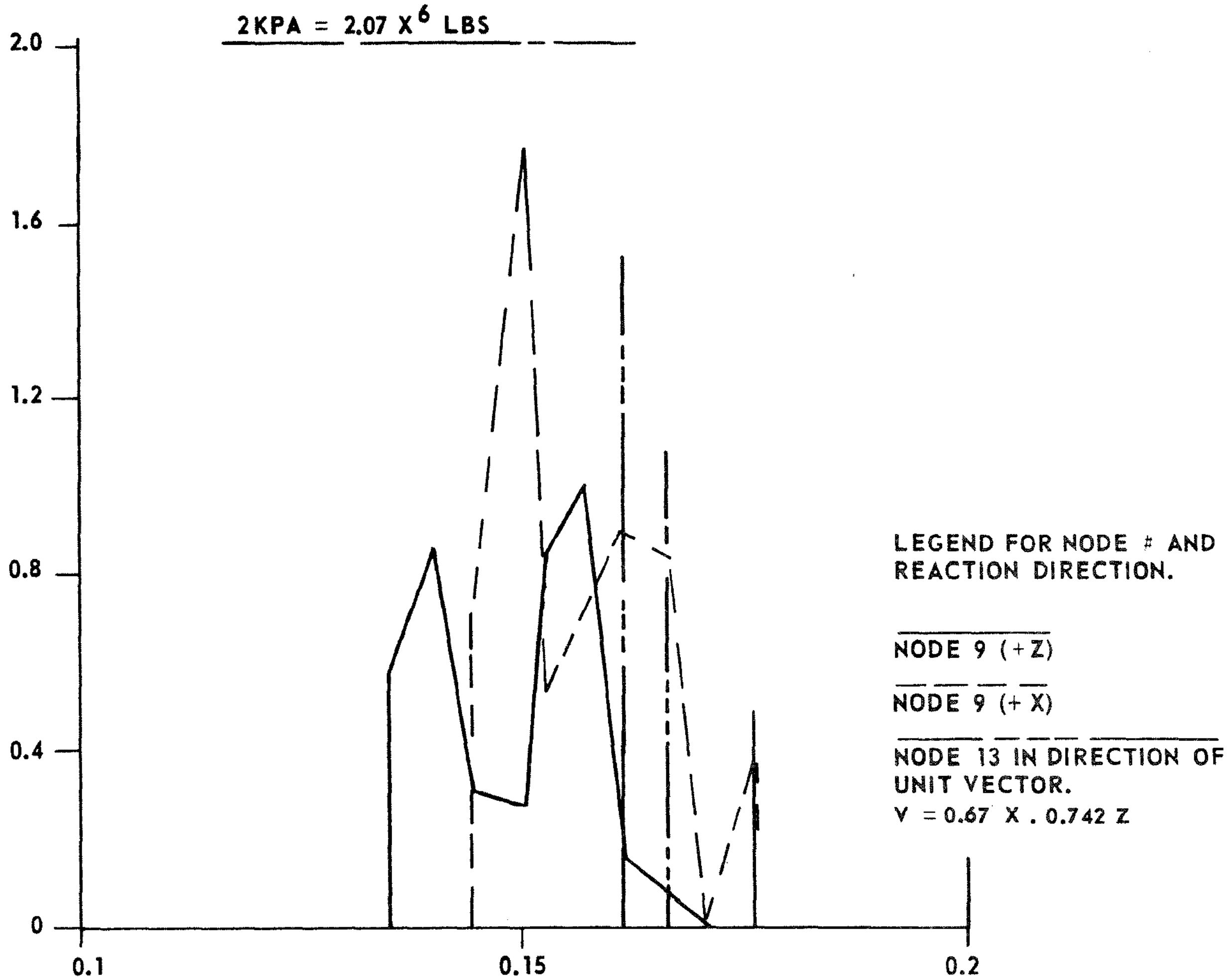
BOILER FEEDWATER LINE INSIDE CONTAINMENT GUILLOTINE BREAK - NODE #7

FIGURE 3.6-55



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

BLOWDOWN DATA FOR MAIN STEAM AND FEEDWATER LINES
FIGURE 3.6-56

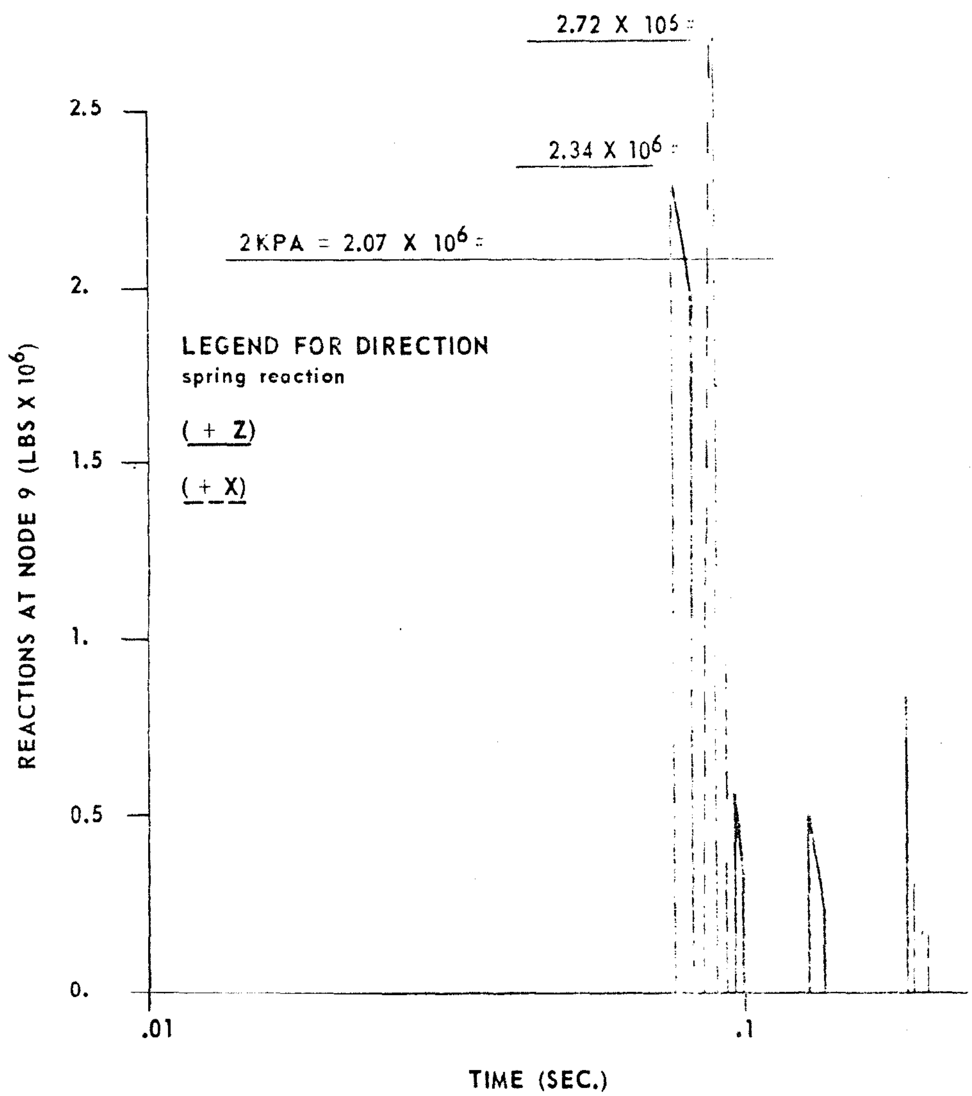


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

MAIN STEAM LINE INSIDE CONTAINMENT (BREAK AT NODE 16) REACTIONS AT PIPE WHIP RESTRAINTS (NODES 13 & 9) VERSUS TIME

FIGURE 3.6-57

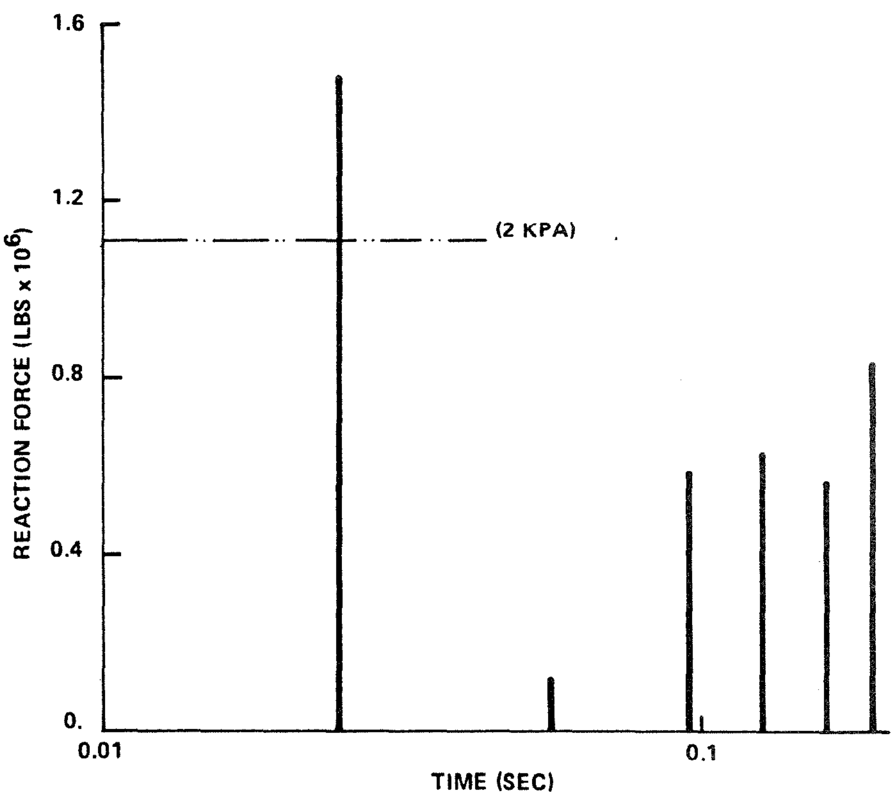


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

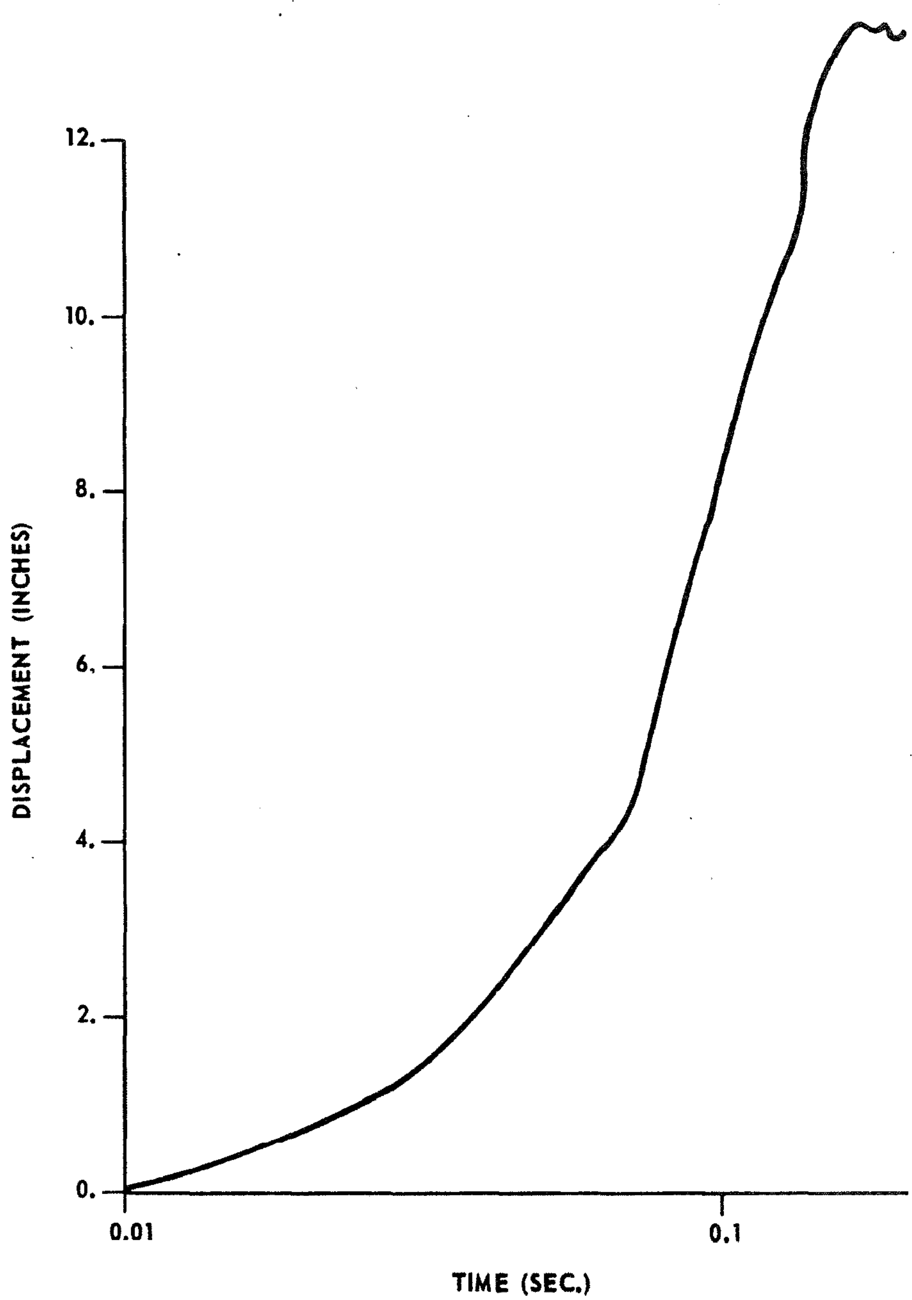
34" MAIN STEAM LINE INSIDE CONTAINMENT (BREAK AT NODE 12) REACTION AT PIPE WHIP RESTRAINT AT NODE 9 IN + X & + Z DIRECTIONS VERSUS TIME

FIGURE 3.6-58



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FW LINE INSIDE CONTAINMENT
REACTION AT PIPE WHIP RESTRAINT
AT NODE 6 VS. TIME
FIGURE 3.6-59

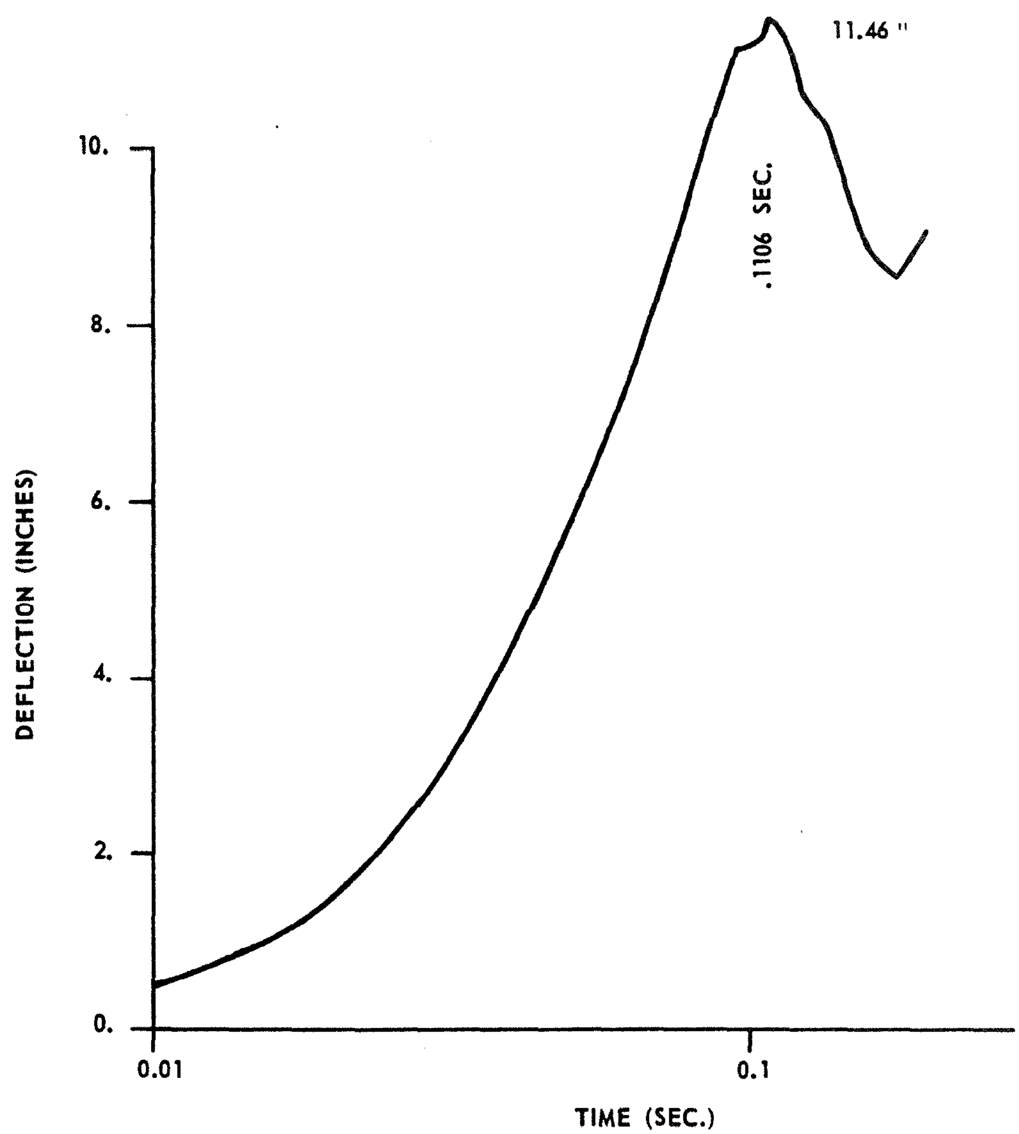


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

MAIN STEAM LINE INSIDE CONTAINMENT BREAK AT NODE 16 DISPLACEMENT OF NODE 16 VERSUS TIME MAX. PLASTIC STRAIN =0.

FIGURE 3.6-60

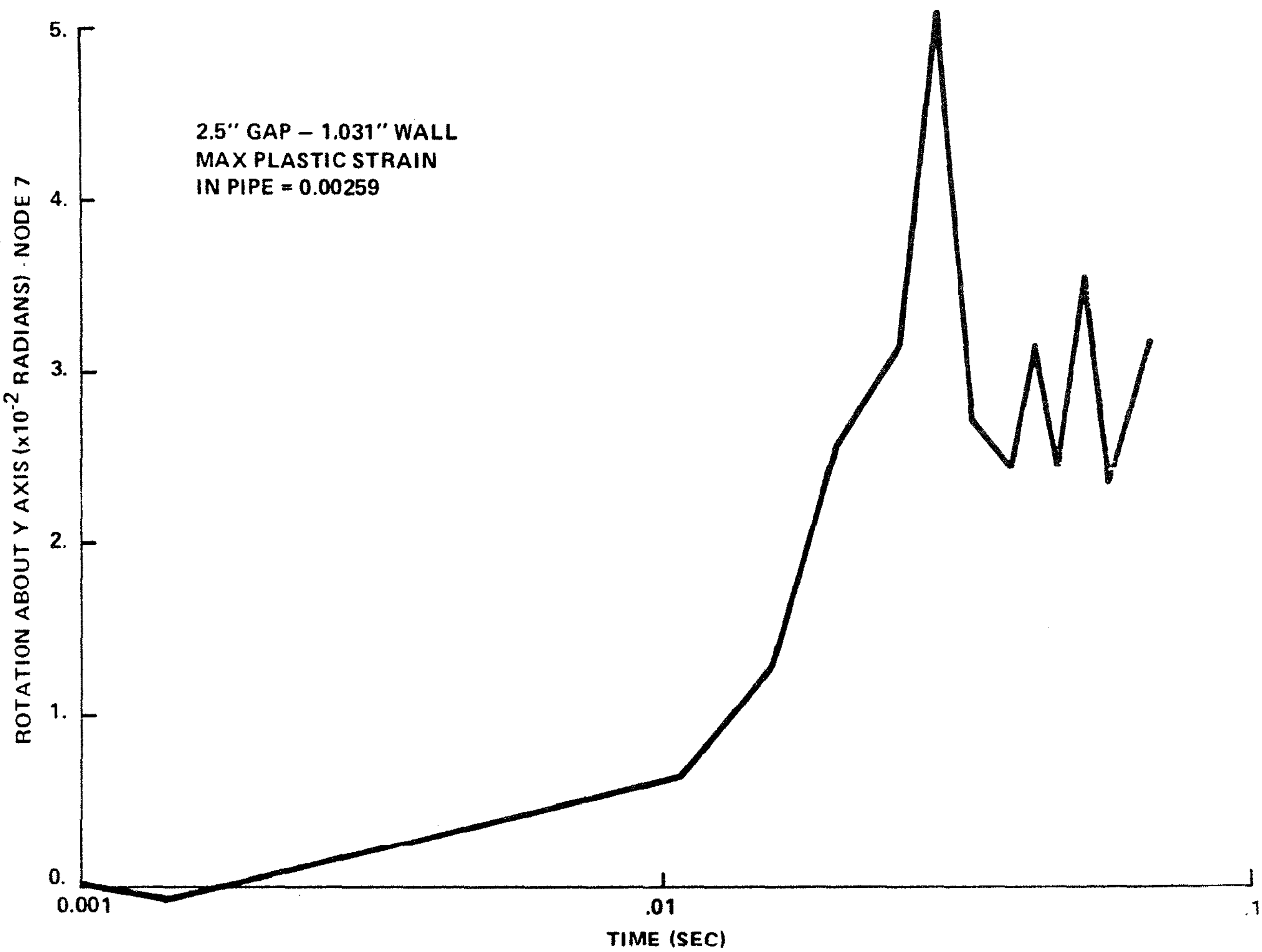


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

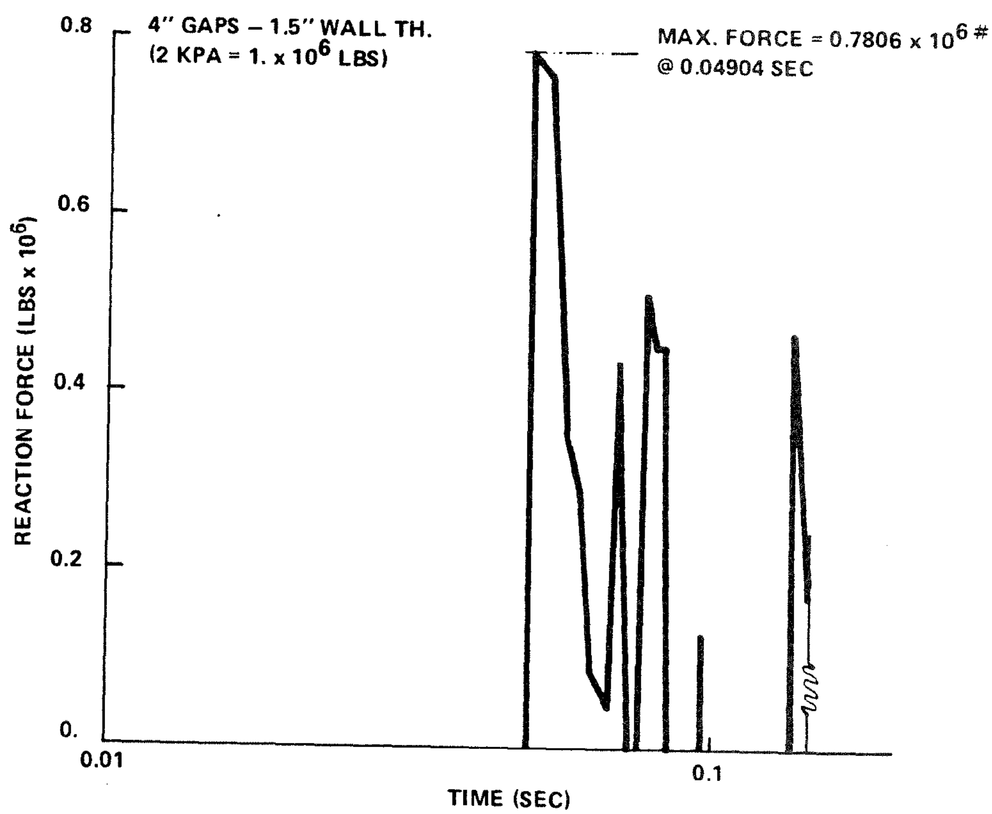
34" MAIN STEAM LINE INSIDE CONTAINMENT (BREAK AT NODE 12) DEFLECTION OF NODE 12 VERSUS TIME MAX. PLASTIC STRAIN = .00762

FIGURE 3.6-61



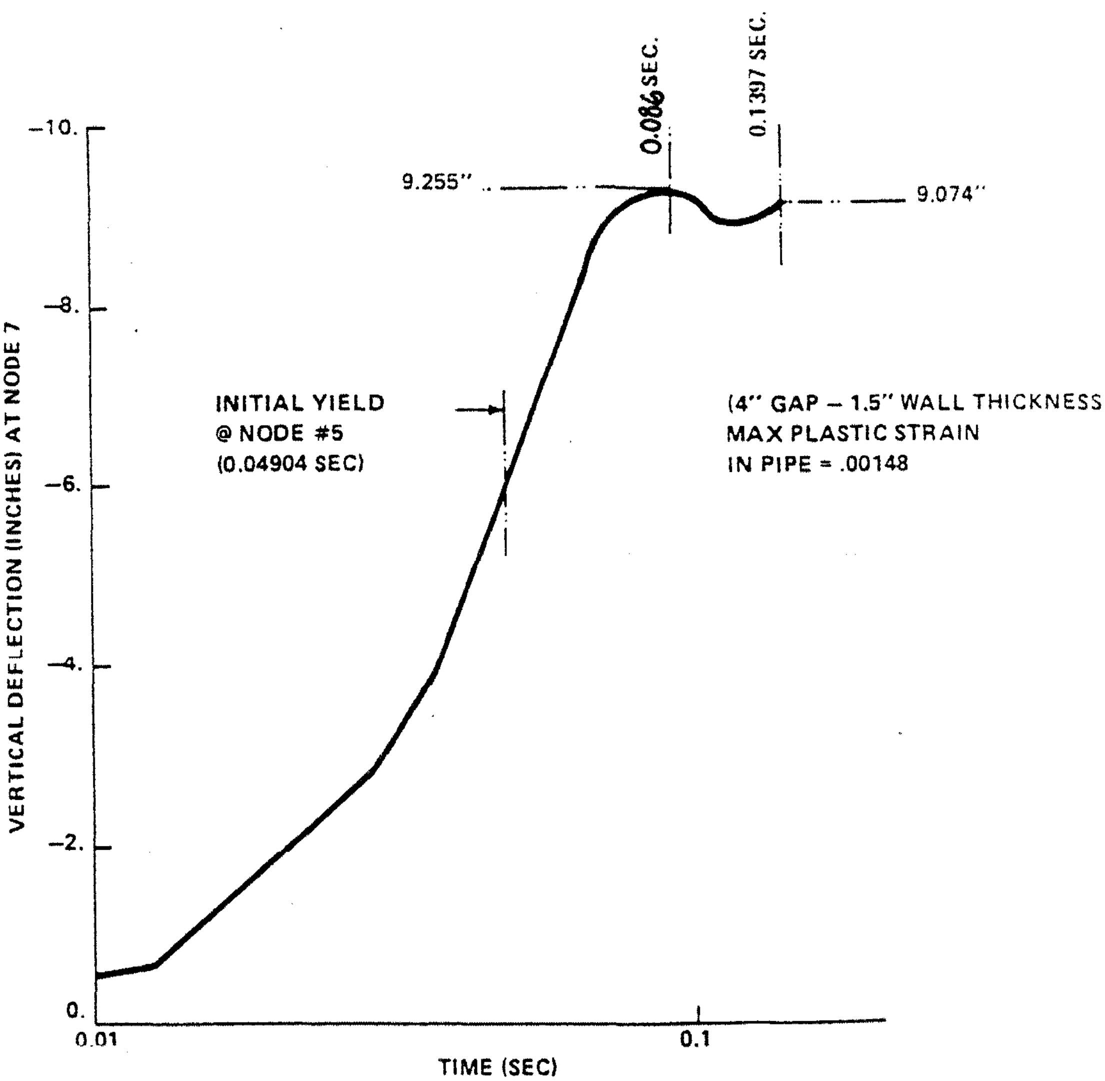
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

20" FW LINE INSIDE CONTAINMENT
ROTATION OF NODE 7 VS. TIME
FIGURE 3.6-62



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

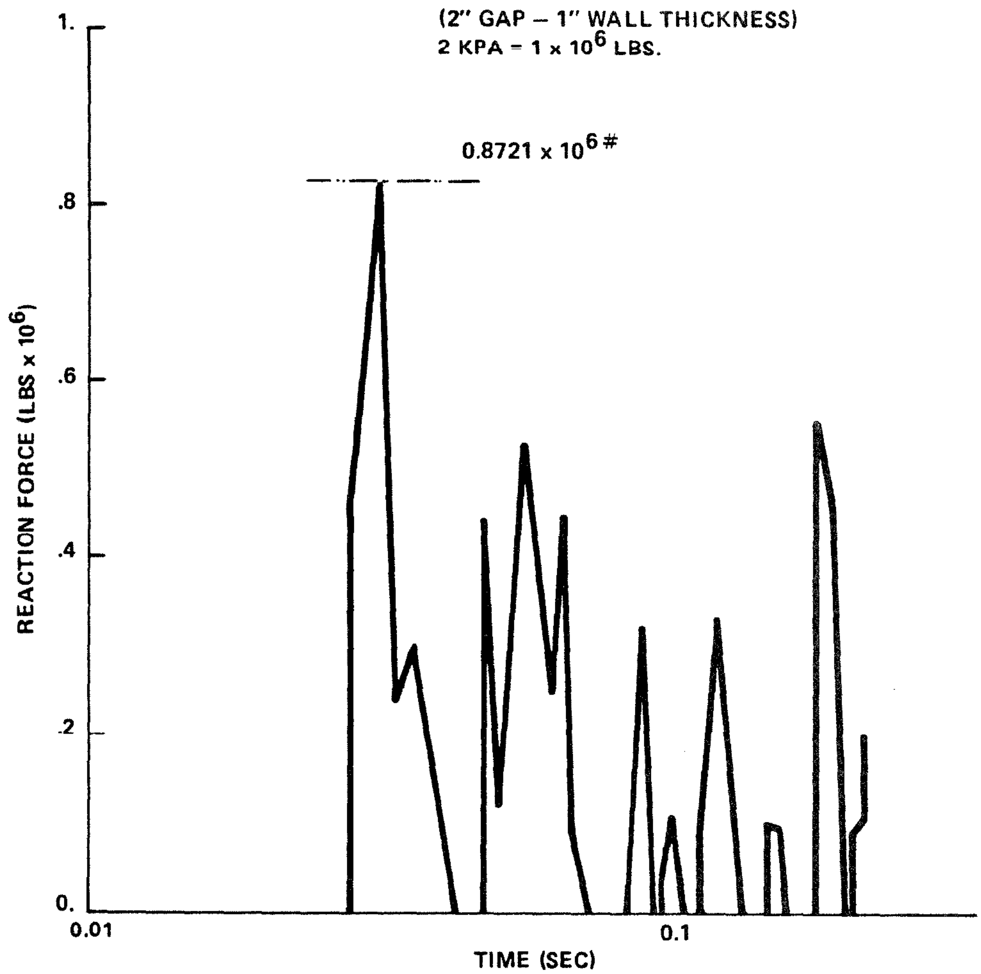
20" FW LINE OUTSIDE CONTAINMENT
REACTIONS AT PIPE WHIP RESTRAINT
AT NODE 5 VS. TIME
FIGURE 3.6-63



Amendment No. 18, (04/01)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

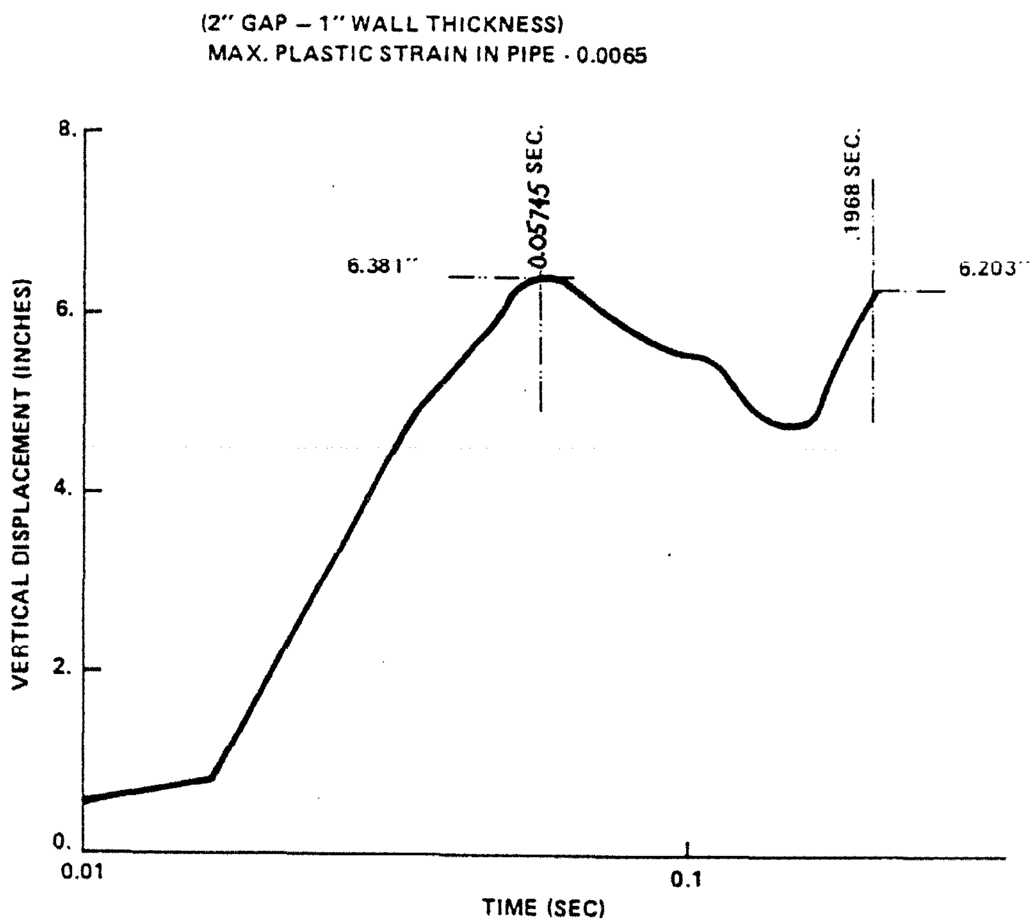
20" FW LINE OUTSIDE CONTAINMENT
DISPLACEMENT AT NODE 7 VS. TIME
FIGURE 3.6-64



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

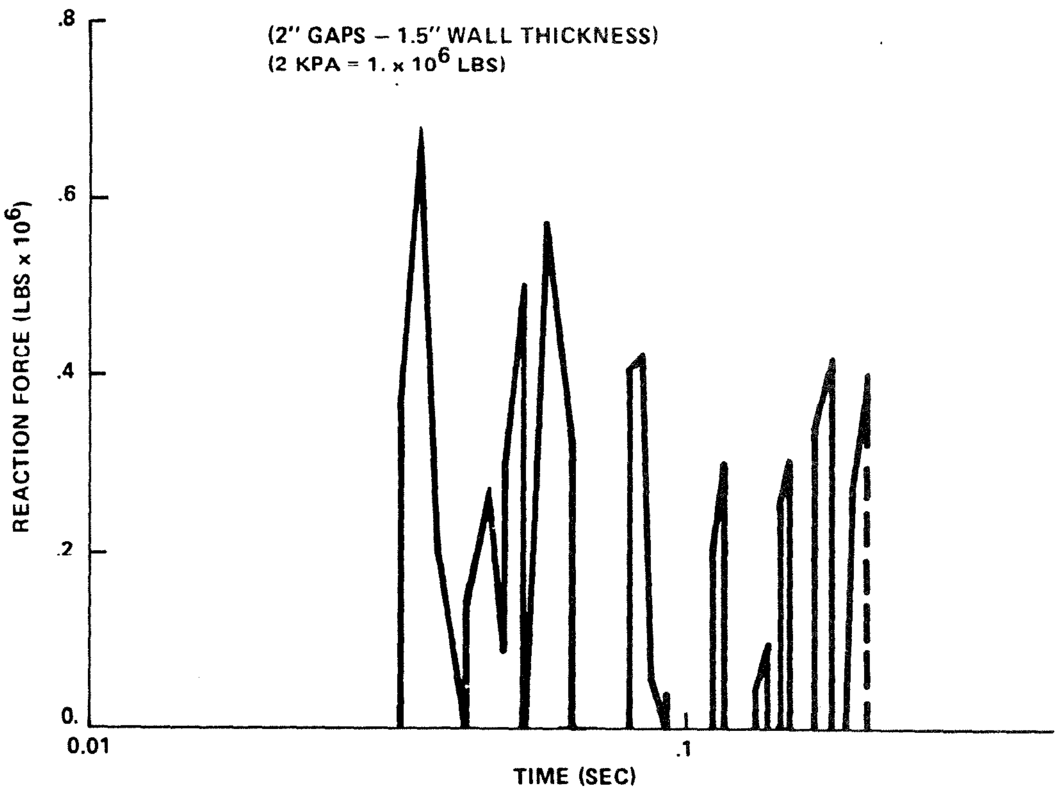
20" FW LINE OUTSIDE CONTAINMENT REACTIONS AT PIPE WHIP RESTRAINT AT NODE 5 VERSUS TIME

FIGURE 3.6-65



Amendment No. 18, (04/01)

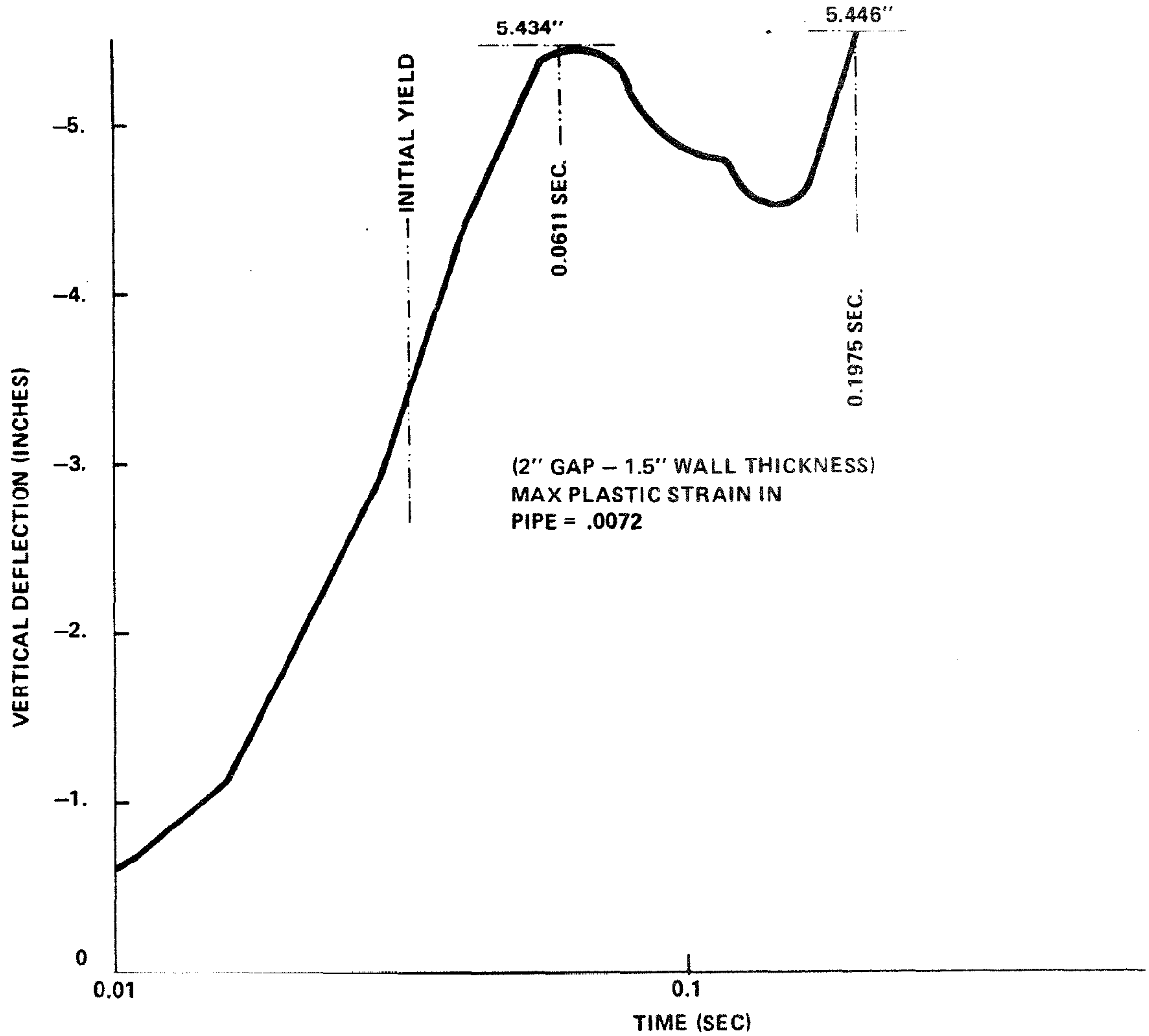
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
20" FW LINE OUTSIDE CONTAINMENT DISPLACEMENT OF NODE 7 VS. TIME FIGURE 3.6-66



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

20" FW LINE OUTSIDE CONTAINMENT REACTIONS AT PIPE WHIP RESTRAINT AT NODE 5 VS. TIME

FIGURE 3.6-67



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

20" FW LINE OUTSIDE CONTAINMENT
DISPLACEMENT OF NODE 7 VS. TIME
FIGURE 3.6-68

3.7 SEISMIC DESIGN

3.7.1 INPUT CRITERIA

3.7.1.1 Design Response Spectra

The design response spectra for the Operating Basis Earthquake (OBE) and Design Basis Earthquake (DBE) are shown on Figures 3.7-1 and 3.7-2 respectively. These spectra were used in the seismic design of safety related structures, systems and components. The design spectra were developed from an envelope of spectra based on vibrating ground motion from several earthquakes, using appropriate amplification factors. The development and justification of the design response spectra (Housner spectra) is presented in Section 2.5.2.

3.7.1.2 Synthetic Time-History Earthquake Records

3.7.1.2.1 Earthquake Simulation Model

Synthetic time-history earthquake records were used in the design of seismic Class I piping systems and equipment.

The ground motions developed were based on a non-stationary, multi-period stochastic earthquake model with variable magnitude and duration developed by Hou (Reference 1) and Cornell (Reference 2). The nonstationary aspect of the model refers to the time variation of ground motion acceleration and the multi-period aspect refers to variations in the relative frequency content of the accelerogram. The stochastic nature of the model results from the random selection of phase shifts in the ground motion periods. In general, N sine terms of different magnitude, amplification and phase shift are added to form the desired ground motion record.

In particular there are four input parameters:

a) Expected duration of earthquake motion

b) Expected maximum ground acceleration

c) Shape of the power spectral density function which defines relative ground motion intensity as a function of period and the shape of the ground motion response spectra

d) The shape of the intensity function which defines the basic shape of the ground motion acceleration record

These input parameters are used to generate synthetic earthquakes as outlined below.

The ordinates of the power spectral density function are normalized so that the area under the density function is unity. These normalized ordinates are then used to calculate normalized or relative amplitudes of N sine terms. A random phase shift between zero and 2π is then calculated and introduced in each sine term. These manipulations provide N stationary time dependent sinusoidal motions with random phase shifts. The normalized intensity function which is an input variable is multiplied by the expected maximum ground acceleration. The resulting time dependent intensity function and the sum of the time dependent stationary sinusoidal motions are then evaluated at given time intervals. These two results are multiplied together at given time intervals to give the amplitudes of the resulting nonstationary ground motion accelerogram at given time intervals.

The velocity responses for various percentages of critical damping are then calculated for each earthquake motion generated. The response is calculated at given periods which are part of the input variables. In addition, average response spectra are calculated for all the records developed from a given set of data.

Sixty-four natural periods of vibration points are used to produce spectra curves from synthetic earthquakes. The breakdown of these periods are indicated in the following:

a) from 0.030 sec to 0.10 sec
12 points are used with 0.005 sec equal interval

b) from 0.10 sec to 0.30 sec
20 points are used with 0.01 sec equal interval

c) from 0.30 sec to 0.50 sec
10 points are used with 0.02 sec equal interval

d) from 0.50 sec to 1.0 sec
10 points are used with 0.05 sec equal interval

e) from 1.0 sec to 2.0 sec
10 points are used with 0.10 sec equal interval

f) 2.5 sec and 3.0 sec.

Response spectra calculated in this manner give irregular response spectra because they are developed from. individual earthquakes. This contrasts with the design response spectra (see Figures 3.7-1 and 3.7-2) where a smooth curve was drawn through the irregular response ordinates to give average spectra.

3.7.1.2.2 Development of Synthetic Earthquake Records

The following are the criteria used for developing the synthetic earthquake records used in the design:

a) Acceleration records for the design earthquake were developed with a magnitude of 0.05 g which have response spectra approximating the Housner spectra. The DBE records were obtained by doubling the magnitude of the OBE records.

b) The input accelerogram peaks to achieve the response spectra were between 0.045 g and 0.055 g for the OBE.

c) The response spectra of the synthetic earthquakes have isolated points falling below the Housner response spectra.

d) The damping factors considered were 0.5 percent, 2 percent and 5 percent.

Based on these criteria, input data were selected to develop the required ground motion accelerograms. A plot of the intensity function is shown on Figure 3.7-3. Since it was necessary to develop earthquakes with response spectra similar to the design response spectra, the power spectral density function was chosen to meet this criteria. This requirement necessitated a trial and error approach.

3.7.1.2.3 Results

Four ground motion acceleration records were developed which meet the proposed criteria. These records have maximum accelerations of 0.049 g 0.049 g, 0.052 g and 0.051 g. The response spectra for each of these records were calculated for 0.5 percent, 2 percent and 5 percent damping. Then the response spectra ordinates of all four records were averaged for each percent of damping. The averaged time-history response spectra are shown in comparison to the design response spectra on Figure 3.7-4. The response spectra calculated in this manner yield a good approximation of the Housner spectra with only isolated points falling below the design response curves. For 0.5 percent critical damping the design response spectrum and the time-history response spectrum of the synthetic earthquakes gave maximum acceleration amplification of 5.0 and 6.6 at periods of 0.2 and 0.27 seconds, respectively.

The requirement of spectra enveloping is satisfied when the averaged spectrum derived from four sets of synthetic earthquake records are used. A comparison between the actual design spectra and the generated average spectra for various damping values are shown in Figure 3.7-4. This shows that the selected synthetic earthquake records are compatible to and in general more conservative than the design spectra.

In the actual design of seismic Class I plant structures, the normal mode method is used based on the actual design spectra. The four synthetic earthquake records are only used to generate floor response spectra for piping or equipment seismic analysis.

The floor response spectra are generated in accordance with the following procedures (Reference 14):

a) Four independent time history analyses are made for all seismic Class I structures using synthetic earthquake records as the input ground motion. The results of these analyses provide four sets of time history responses of each floor level where floor response spectra are to be generated.

b) Four floor response spectra with appropriate damping values are generated using standard numerical integration methods based on four independent time history floor responses.

c) The final floor response spectra to be used for piping and equipment seismic analyses and design envelop all four floor spectra as described in item (b).

The method proposed above is adequate and sufficient to assure a reasonable and conservative design of plant structures, piping and equipment.

3.7.1.3 Critical Damping Values

Values of critical damping used in the seismic analysis are as follows for the various types of structural members:

PERCENT CRITICAL DAMPING

	OBE (0.05g ground surface acceleration)	DBE (0.10g ground surface acceleration)
Welded steel plate assemblies	1	1
Steel containment vessel	2	2
Welded steel framed structures	2	2
Bolted or riveted steel framed structures	2.5	2.5
Reinforced concrete equipment supports	2	5
Reinforced concrete frames and buildings	2	5
Steel piping[1]	0.5	0.5
Soil	10	10

3.7.1.4 Site Dependence of Analyses

The development of the seismic design response spectra was based on expected response of the subsurface materials at the site to seismic excitation. The analysis includes amplification of the seismic accelerations from bedrock to ground level due to the response of the overlying soil layers. The effects of site related parameters on the seismic response spectra are discussed fully in Section 2.5.2.

[1] For analysis of piping, damping values specified by ASME Code Case N-411 may be applied. Refer to Section 3.9.2.3 (pages 3.9.23 and 3.9.23a) of this UFSAR for details.

3.7.1.5 Soil-Structure Interaction

All structures are soil supported. Due to the rigid and massive behavior of all seismic Class I structures and the relatively soft soil characteristics, it was anticipated that considerable rocking and translating motions of structures may take place during a severe earthquake at the site. To include these motions in the seismic analysis, it was considered appropriate to model the soil into rotational and translational springs to allow for these additional degrees of freedom. The spring constants were calculated using the following formulas from Reference 3.

a) For a rectangular foundation

$$K_\phi = \frac{G}{1-\mu}\beta_\phi BL^2$$

$$K_x = 2(1+\mu)G\beta_x(BL)^{1/2}$$

$$K_z = \frac{G\beta_z}{1-\mu}(BL)^{1/2}$$

b) For circular foundation

$$K_\phi = \frac{8Gr^3}{3(1-\mu)}$$

$$K_x = \frac{32(1-\mu)Gr}{7-8\mu}$$

$$K_z = \frac{G}{1-\mu}\beta_z\sqrt{2rH}$$

Where $K\phi$ is the rocking soil spring constant, K_x is the horizontal soil spring constant, and K_z is the vertical soil spring constant

G = shear modulus of soil
μ = Poisson's ratio of soil
B = width of rectangular foundation
L = length of rectangular foundation
H = depth of foundation
r = radius of circular foundation
and $\beta\phi$, β_x, β_z site constants dependent on B/L ratio

From laboratory soil testing and analyses (see Section 2.5.4.4) the proper Young's modulus used for the calculation of soil spring constants for all seismic Class I structures was determined as follows:

Reactor Building	E=40,000 psi
Reactor Auxiliary Building	E=40,000 psi
Fuel Handling Building	E=35,000 psi
Diesel Generator Building	E=30,000 psi
Intake Structure	E=40,000 psi

For all above cases, the Poisson's ratio (μ) for soil is 0.25.

As discussed in Section 2.5.4.4 (b), Poisson's ratio varies with strain. For the early stages of a first
loading of a dense sand, when intermediate strain levels are developed and particle rearrangements are important, "μ" typically has values of about 0.25 as selected above and is consistent with the anticipated building strains of 10^{-3} to 10^{-4} in/in.

Preliminary analysis of the reactor building was performed using a range of soil moduli of 10,000 to 250,000 psi. This range became unrealistic for piping system design and therefore the additional laboratory testing described in Section 2.5.4.4 was conducted to more closely define the soils modulus. The values obtained from this additional testing were used in the final design of Class I structures.

In the actual analysis, in order to include any uncertainties of the selected soil modulus on the structural responses, a range of soil moduli within ±20 percent of the selection values were used.

Table 3.7-1A provides a tabulation of all soil-supported seismic Class I structures and the depth of each of the various soil layers to the bottom of the excavation line at elevation -60 ft.

The effects of soil-structure interaction for all seismic Class I structures are considered in the seismic analysis by means of providing equivalent foundation and translational springs based on the theory of rigid plates on elastic half space. To include the effects of any uncertainties of foundation soil engineering properties, a parametric study is made to vary soil properties by ±20 percent. The maximum responses resulting from the parametric study are used for the actual design.

The appropriateness of the methods used in calculating soil-structure interaction are based on the following:

a) Soil properties, such as shear modulus and Poisson's ratio were determined both from laboratory and field tests. The shear modulus versus strain relationship is established as shown in Figure 2.5-32.

b) The shear modulus used for the calculation of equivalent foundation spring constants is consistent with the anticipated soil strain during a DBE. Section 2.5.4.4 addresses the procedures used in determining the anticipated soil strain levels during earthquakes.

c) The maximum building embedment depth from plant grade is approximately 43 ft for the reactor building. The embedment is approximately 25 percent of the reactor building diameter and less than 20 percent of the total building height. In general this type of embedment could be neglected in soil-structure interaction calculations without a significant effect on the structural responses when using the concept of a rigid body resting on an elastic half space. However, in the actual calculation, the shallow embedment effects are included by providing a side spring at approximately the middle of the embedment.

d) To verify that the model selected and method used for the soil-structure interaction effects is appropriate, a set of calculations were made using different values for the side spring constant. The insignificant differences obtained by varying the spring constant indicates that the reactor building embedment has very little effect on the overall structural response. Refer to Table 3.7-1B.

3.7.2 SEISMIC SYSTEM ANALYSIS

This section includes discussion of seismic analysis of all seismic Class I structures. Seismic analysis of seismic Class I piping system including reactor coolant system is discussed in Section 3.7.3.

3.7.2.1 Method of Analysis

3.7.2.1.1 Mathematical Models

For the seismic analysis of all Class I structures, conventional lumped mass mathematical models were selected to represent each structure. In this model, the structure is represented by a cantilever beam with masses lumped at selected elevations simulating floor weights, walls, columns and major equipment. The cantilever beam connecting those lumped masses is assumed weightless and elastic representing the stiffness of walls or columns between the lumped mass points. The foundation mat supporting the cantilever beam is considered as a rigid body and is supported by rotational and translational springs simulating soil-structure interaction.

For the seismic analysis in the vertical direction, mathematical models were developed using similar lumped mass principles. However, since the major interest in this case is focused at the middle of a floor bay or at column floor junctures, appendages representing floor bay behavior are added to the cantilever beam resulting in a more complex model.

Equivalent soil springs as described in Section 3.7.1.5 and critical damping values as described in Section 3.7.1.3 were used in the analysis. Details of the mathematical models used for the various seismic Class I structures are discussed below:

a) Reactor Building

For structural responses in the horizontal direction, the mathematical model consists of three independent cantilever beams representing the steel containment vessel, shield building and internal structure respectively. Masses are lumped at ten selected locations for the steel containment and the shield structure and are lumped at four locations for the internal structure. These three cantilever beams are supported by the rigid foundation mat approximately 43.5 feet in depth. Rotational and translational springs are connected to the mat simulating soil-structure interactions. This model is shown in Figure 3.7-5. Table 3.7-1 describes mass and stiffness characteristics of this model.

For structural responses in the vertical direction, the model consists of three cantilever beams. Five mass points are used to represent both the steel containment vessel and the shield building since in the vertical case, less variation of the structural responses is anticipated. For the internal structure, four mass points are used. This model is shown on Figure 3.7-6. Table 3.7-2 describes mass and stiffness characteristics of this model.

b) Reactor Auxiliary Building

For structural responses in horizontal directions, two mathematical models were used corresponding to the N-S (long) and the E-W (short) direction, of the building. Each model consists of a single cantilever beam with four lumped masses. The cantilever beam is supported on the rigid foundation mat which in turn is supported by rotational and translational springs simulating the soil-structure interaction. The model is shown in Figure 3.7-7. Table 3.7-3 describes the mass and stiffness characteristics of these two models.

For structural responses in the vertical direction, the complexity of the building makes it impossible to model the whole structure into a reasonable simple lumped mass mathematical model. It was decided that the coupled motion between adjacent floor bays, which is anticipated to be small, be neglected in the model thus allowing establishment of a relatively simple model and yet sufficient to yield reasonable results. This model as shown in Figure 3.7-8 consists of a single cantilever beam which actually represents the total stiffness of all vertical structural elements of the building. At four different floor elevations, appendages representing the behavior of floor bays were attached to the cantilever beam which is supported by the rigid foundation mat. The foundation mat is supported by vertical soil springs simulating soil-structure interaction. Table 3.7-4 describes the mass and stiffness characteristics of this model.

c) Fuel Handling Building

For structural responses in horizontal directions, two models were used corresponding to the N-S (long) and the E-W (short) directions of the building. Each model consists of a cantilever beam with three lumped masses. The cantilever beam is supported by the rigid foundation mat which in turn is supported by rotational and translational springs simulating soil structure interactions. The model is shown in Figure 3.7-9. Table 3.7-5 describes the mass and stiffness characteristics of these two models.

d) Intake Structure

For structural responses in the horizontal directions, two mathematical models were used corresponding to each direction of the structure. The model consists of a cantilever beam with masses lumped at three selected places. Since the intake structure is essentially buried underground, a lateral spring is used at each lumped mass to simulate the interactions. The cantilever beam is supported on the rigid foundation mat which in turn is supported by rotational and translational springs simulating soil structure interactions. The model is shown in Figure 3.7-10. Table 3.7-6 describes the mass and stiffness characteristics of this model.

For structural responses in the vertical direction, the mathematical model consists of a cantilever beam with three lumped mass points. One appendage is attached to the top most lumped mass representing the behavior of the top deck of the intake structure. The cantilever beam is supported on the rigid foundation mat which in turn is supported by the vertical soil spring simulating soil-structure interactions. The model is shown in Figure 3.7-10. Table 3.7-6 describes the mass and stiffness characteristics of this model.

e) Diesel Generator Building

For structural responses in the horizontal directions, two mathematical models were used corresponding to each direction of the building. The model as shown in Figure 3.7-11 consists of three cantilever beams; one representing the structure and two representing the diesel generators and their foundations. All three cantilever beams are supported on the foundation mat which in turn is supported by rotational and translational springs simulating soil-structure interaction. Table 3.7-7 describes the mass and stiffness characteristics of the horizontal model.

For responses in the vertical direction, the model consists of three cantilever beams with one appendage attached to the top most mass of the cantilever beam No. 1 to represent the behavior of the roof of the building. This model is shown in Figure 3.7-11.' Table 3.7-7 describes the mass and stiffness characteristics of this model.

3.7.2.1.2 Equations of Motion

Once the mathematical model is established, the motion of each lumped mass under any external excitation may be written in the matrix form as follows:

$$[M]\{\ddot{\Delta}\} + [K]\{\Delta\} = \{F\} \qquad (1)$$

Where: $[M]$ = Square mass matrix

$[K]$ = Square matrix of stiffness coefficients including the shear and bending deformations

$\{\ddot{\Delta}\}$ = Column matrix of acceleration vectors

$\{\Delta\}$ = Column matrix of lateral displacement and joint rotation vectors

$\{F\}$ = Column matrix of external load vectors

The stiffness matrix $[K]$ is formulated by computing the stiffness coefficients for each joint of the original structure and assembling them in the proper sequence to form the complete square matrix. In the computation of the stiffness matrix, it is assumed that all joints at the same level have the same displacements (i.e., translations and rotations).

In the above equations of motion, the damping terms are left out intentionally. This is due to the fact that the damped natural frequency is almost the same as the undamped natural frequency for the system with reasonable structural damping factors (for 10 percent critical damping, d = 0.995).

3.7.2.1.3 Natural Frequency and Mode Shapes

In calculating the natural frequencies and the mode shapes, the external load matrix in eq (1) is set to zero, the displacement vector $\{\Delta\}$ is assumed to take the form of simple harmonic motion, or

$$\{\Delta\} = \{\phi\} \operatorname{Sin} \omega t \tag{2}$$

Where $\{\phi\}$ = Relative amplitude of mode shape vector

ω = Natural frequency of vibration.

After substituting and simplifying, the equations of motion are reduced to the following form:

$$[K]^{-1}[M]\{\phi\} = \frac{1}{\omega^2}\{\phi\} \tag{3}$$

Solution to this eigenvalue problem exists only for particular values of ω which correspond to the natural frequencies of vibration of the structure. Eq (3) is solved by iteration techniques to obtain values *of w* and their corresponding mode shape vectors $\{\phi\}$.

3.7.2.1.4 Modal Analysis

After all natural frequencies and their mode shapes are determined, the method of modal analysis is employed to calculate the structural responses. This method actually simplifies the analysis of a multidegree of freedom system to the analysis of several equivalent single degree systems, one corresponding to each normal mode. The governing equation of motion is shown in the following:

$$\ddot{A}_n + 2B_n\dot{A}_n + \omega_n{}^2 A_n = \frac{-\ddot{Y}_{so}\, f_a(t) \sum_{x=1}^{N} M_x \phi_{xn}}{\sum_{x=1}^{N} M_x \phi_{xn}{}^2} \tag{4}$$

in which

A_n = Displacement of any one arbitrarily selected mass (Usually the topmost mass) of the nth mode

B_n = Damping coefficient = $\lambda_n\ \omega_n$

λ_n = Percentage of critical damping of the nth mode

ω_n = Natural frequency of the nth mode

$\ddot{Y}_{so}$ = Maximum ground acceleration

$f_a(t)$ = Time function of ground motion

M_x = Mass at the xth level

ϕ_{xn} = Normalized displacement of the mass. M_x of the nth mode

If the two summations on the right-hand side of the eq (4) are denoted by P_n, which is defined as the modal participation factor of the nth mode, then

$$\ddot{A}_n + 2\,\mathbf{B}_n\dot{A}_n + \omega_n A_n = -P_n\ \ddot{Y}_{so}\ f_a(t) \tag{5}$$

Since the values of B_n, ω_n and P_n are already known for each normal mode, eqs (5) which are actually "n" independent equations, can be solved separately and their solutions are:

$$A_n(t) = -\frac{P_n\ddot{Y}_{so}}{\omega_n}\int_o^t f_a(t)e^{-\lambda n\omega n(t-\tau)}\ \mathrm{Sin}\ \omega_n(t-\tau)d\tau \tag{6}$$

The maximum values of eq. (6) are:

$$A_n(t)_{max} = -P_n\left\{\frac{\ddot{Y}_{so}}{\omega_n}\int_o^t f_a(t)e^{-\lambda n\ \omega n(t-\tau)}\ \mathrm{Sin}\ \omega_n(t-\tau)\,d\tau\right\}_M \tag{7}$$

Since $Y_{xn}(t) = \phi_{xn} A_n(t)$, therefore

$$Y_{xn}(t)_{max} = -P_n\,\phi_{xn}\left\{\frac{\ddot{Y}_{so}}{\omega_n}\int_o^t f_a(t)e^{-\lambda n\ \omega n(t-\tau)}\ \mathrm{Sin}\ \omega_n(t-\tau)\,d\tau\right\}_M$$

$$= -P_n\ \phi_{xn}\ S_{dn} \tag{8}$$

where S_{dn} indicates the quality in the bracket. Finally the total displacement is the summation of the displacement of each normal mode, that is:

$$Y_x(t)_{max} = -\sum_{n=1}^{N} P_n\,\phi_{xn}\ S_{dn} \tag{9}$$

Eq (9) gives the upper limit of the displacements of any mass. However, as we can reasonably assume that all the maximum displacements of all normal modes do not necessarily occur at the same time, therefore, for the purpose of design, the root-mean square-method is adopted from the statistical point of view, thus:

$$Y_x(t)_{max} = \left[\sum (P_n \phi_{xn} S_{dn})^2 \right]^{1/2} \qquad (10)$$

3.7.2.1.5 Structural Responses

Knowing the function $\ddot{Y}_{so} f_a(t)$, or the record of the design earthquake, the value of S_d can be calculated and plotted into a curve using the natural period of vibration as abscissa and the maximum displacement as the ordinate. This curve is known as the displacement response spectrum. To construct the velocity response spectrum, it is found that the spectral displacement S_d is directly related to the spectral velocity S_v by the natural frequency ω, or $S_v = \omega S_d$. Similarly, the spectral acceleration $S_a = \omega S_v = \omega^2 S_d$. In the design of the structure the response spectra employed were the spectra given in Section 3.7.1.1. The structural responses are then calculated through the,following operations.

$$M_n^* = \frac{\left(\sum_{x=1}^{N} \phi_{xn} M_x \right)^2}{\sum_{x=1}^{N} \phi_{xn}^2 M_x}$$

$$V_n = M_n^* \omega_n S_{vn}$$

$$F_{xn} = V_n \left[\frac{\phi_{xn} M_x}{\sum_{x=1}^{N} \phi_{xn} M_x} \right]$$

$$\Delta_{xn} = [K]^{-1} (F_{xn})$$

Where M_n^* = Effective mass in nth mode

M_x = Mass. concentrated at xth level

ϕ_{xn} = Normalized displacement of M_x in the nth mode

V_n = Base shear in nth mode

Δ_{xn} = Deflection of the xth level in nth mode

F_{xn} = Inertia force at the xth level in nth mode

S_{vn} = Spectra velocity in nth mode

ω_n = Natural frequency of nth mode

The modal response values for each mode computed above are combined by the root-mean-square-method to determine the total responses of the
structure, thus

$$(F_x) = (\sum_{n=1}^{N} F_{xn}^2)^{1/2} \quad (11)$$

$$(\Delta_x) = (\sum_{n=1}^{N} \Delta_{xn}^2)^{1/2} \quad (12)$$

3.7.2.1.6 Analysis Procedures for Composite Damping

The procedure used to calculate the composite damping ratio for the natural modes of a structure having composite materials or a substructure with different damping ratios is as follows:

$$D_n = \frac{\sum_{i=1}^{m} d_i \; S_{ni}}{S_n}$$

where:

D_n = percentage of the critical damping ratio for the nth mode

d_i = percentage of the material damping ratio for the ith structural component

S_{ni} = strain energy of the i th structural component in the nth mode

S_n = total strain energy of the structure in the n th mode

m = number of structural components

The composite damping ratio as calculated by the above expression is based on the method of strain energy proportioning.

3.7.2.2 Vertical Analysis

A multi-mass dynamic analysis procedure was used for the vertical response loading for the seismic design of buildings and floors. The methods of analysis used for the vertical dynamic analysis are the same as those described in Section 3.7.2.1.

The equivalent vertical static loads are combined with the vertical dead and live loads and other factored loads as discussed in Section 3.8 for the design of the structural markers affected.

3.7.2.3 Torsional Modes of Vibration

A static factor has been employed for torsional modes of vibration in the seismic analysis of seismic Class I structures. Design calculations for the torsional stress in the exterior walls of the buildings based on uniform building code indicates that the structure is designed adequately for torsional loading. Additional substantial torsion resisting capacity is inherent in the structures due to interior reinforced concrete walls. Hence use of a static factor for torsional modes of vibration in the seismic analysis of the buildings in lieu of a combined vertical, horizontal and torsional multimass system dynamic analysis is considered to be adequate.

3.7.2.4 Comparison of Modal Analysis and Time History Methods

In order to provide a check on the seismic analysis of Class I structures, an analysis of the reactor auxiliary building using both the modal analysis response spectra method and time history method was conducted. Table 3.7-8 gives the response at selected points within the reactor auxiliary building for both these methods. As can be seen, the response spectra method results in higher responses at certain points for a DBE in the N-S direction, while the time history method gives higher responses for the E-W direction.

3.7.2.5 Overturning Moments

The horizontal seismic response loads acting at their corresponding mass point elevations, determined from the dynamic analysis described in Section 3.7.2.1, are used in computing the overturning moments about the base of the structure.

Vertical earthquake effects are considered by deducting from the dead load righting moments the vertical response loads determined by the vertical dynamic analysis. Buoyancy, where it is present, is also considered in the summation of moments.

Where structures are embedded in soil strata, resisting soil pressures acting as righting moments are not included in the net overturning moment. However, where dynamic effects of soil strata contribute to overturning moments these horizontal loads are considered by including this additional overturning in the final summation. The resulting soil reactions, with the appropriate combination of vertical response loads added to give the maximum effects, are compared against the allowable dynamic soil pressures to assure compliance to the criteria.

3.7.2.6 Results of Analyses

Five analyses were made for all seismic Class I structures both in the horizontal and in the vertical directions. Each analysis was made using one particular set of soil springs calculated from one particular soil modulus. Other properties of the structure were kept the same for all five analyses. The design concrete strength for the reactor building was 4000 psi. The concrete Young's modulus was 481,500 ksf. For all other structures the design concrete strength was 3000 psi and Young's modulus was 422,000 ksf. A Poisson's ratio of 0.17 was used for all structures. The structural responses and natural periods of vibration for all seismic Class I structures are presented in Tables 3.7-9 through 3.7-21.

3.7.2.7 Computer Programs Utilized for Structural and Seismic Analyses

3.7.2.7.1 Description

The following computer programs have been used in structural and seismic analyses to determine stresses and deformations of seismic Class I structures. A brief description of each program and the extent of its use are given below:

a) SHELLS Program

This program uses techniques of finite difference method to determine stresses and deformations of a shell structure in the form of a surface of revolution about an axis. Loadings on the structure can be either axial, symmetrical or arbitrary. Arbitrary loads are handled by Fourier expansion techniques. The SHELLS program was used to perform static and thermal stress analysis of the dome and cylinder portion of the reinforced concrete shield building structure.

b) SOLIDS II Program

The SOLIDS II program utilizes a finite element method as applied to solids with an axis of structural symmetry and subjected to Fourier expansions of thermal, body force and surface traction loadings. This program was used to perform stress analysis of the dish shape foundation mat of the shield building structure.

c) SAMIS Program

The SAMIS program employs the finite element stiffness method using both beam and plate-type elements to solve general structural problems. This program was used to perform stress analyses for both the reactor building internal structure and the shield building structure in the areas of large openings.

d) FROA-2034 Program

This program handles the dynamic analysis of lump-mass-spring type models. It provides results of natural periods of vibration; mode shapes, participation factors and structural responses. Both methods of time history and response spectrum calculation can be specified. This program was used for all seismic analyses of seismic Class I structures.

e) SHAKE-2034 Program

This program calculates maximum response of a single degree of freedom system subjected to base time motions. By specifying frequency and damping values, a family of spectra curves are provided by the program. This program was used to calculate all floor response spectra curves.

f) RFRM-117 Program

This program analyzes a two dimensional single or multi-story frame under vertical or horizontal loads. This is accomplished by using a stiffness matrix approach with a Gaussian elimination method. This program was used for frame analysis of all seismic Class I structures.

g) EBS/NASTRAN

EBS/NASTRAN is an enhanced NASTRAN program developed by EBASCO which has the ability to perform concrete cracking analysis. This feature incorporates a special plate element which consists of a user specified number of layers, each having a different proportion of steel to concrete area, representing the presence of reinforcing steel. Each layer will crack or re-close according to the stress-strain relationships of the concrete and steel. Thus a cracking pattern and stress redistribution can be determined. This program was used to analyze the spent fuel pool structure for increased load, seismic and thermal loads resulting from the 1987/1988 spent fuel rack replacement.

h) POSBUKF

POSBUKF is a program developed by EBASCO to examine the elastic post-buckling behavior of a flat plate subjected to thermal and lateral loading using an energy method approach. The program determines the deflected shape of a buckled plate by minimization of potential energy, and from this calculates plate stresses utilizing strain-displacement and stress-strain relationships for the particular case under study.

This program was used for the spent fuel pool liner buckling analysis for increased thermal and strain induced loads resulting from the 1987/1988 spent fuel rack replacement.

3.7.2.7.2 Program Applicability and Validity

The SHELLS, SOLIDS II, and SAMIS programs are structural analysis computer programs originally developed for the aerospace industry and later modified to suit general applications. At the time when structural analyses for St. Lucie Unit 1 were performed, these programs were the properties of the Service Bureau Corporation, a subsidiary of IBM. The dependability of these programs have long been established through the wide use by many industries. These programs were written in FORTRAN language and operated on an IBM 360/65 machine.

The FROA and SHAKE programs were developed by EBASCO for the purpose of performing seismic analysis of structures. These programs were written using ALGOL language and operated on a Burrough 6600 machine. A comparison between the results of FROA and the STARDYNE program is shown in Table 3.7-21A. STARDYNE is a well known and proven computer program existing in the public domain. No comparison effect is made for the SHAKE program since the logic used in SHAKE is a part of the FROA program.

The RFRM program is also an EBASCO program written in FORTRAN language and operated on a Burrough 6600 machine. Due to the relatively simple nature of the program, comparison of results were made by solving several sample problems whose answers were already known. The minor differences as shown were from the secondary effect of column shortening which is considered by the RFRM program.

As discussed above, the SHELLS, SOLIDS II and SAMIS programs are proven programs existing in the public domain and therefore no comparison of results with other programs is presented. For
EBASCO programs, FROA and RFRM, several comparisons were made and are presented in Tables 3.7-21A and 3.7-21B.

3.7.3 SEISMIC SUBSYSTEM ANALYSIS

3.7.3.1 Seismic Input Data

The procedure used to account for the number of earthquake cycles during one seismic event includes consideration of the number of significant motion peaks expected to occur during the event. The number of significant motion peaks during one seismic event would be expected to be equivalent in severity to no more than 40 full load cycles about a mean value of zero and with an amplitude equal to the maximum response produced during the entire event. Based upon this consideration and the assumption that seismic events equivalent to five Operating Bases Earthquakes will occur during the life of the plant, Category I systems, components and equipment are designed for a total of 200 full load cycles.

The analysis used to establish loadings for seismic design of components and equipment depends upon the complexity of the structural model required to define the dynamic response. In each case, the structural (or mathematical) model used will provide sufficient detail to reflect the contribution of all significant dynamic modes of response- under seismic excitation.

For seismic analysis of seismic Class I equipment and piping systems, building floor response spectra were developed using synthetic earthquake time history records as described in Section 3.7.1. The time history records were developed using a 20 second duration of earthquake motion. The floor response spectra are shown on Figure 3.7-12 through 3.7-24. Generally the floor response spectra exhibit two major peaks corresponding to the first and second modes of vibration. To preclude a resonant condition at these peak accelerations, the design period of vibration of piping systems has been limited to no more than 70 percent of the second mode period of vibration which is the lowest peak shown on the spectra.

3.7.3.2 Seismic Analysis - Reactor Coolant System

3.7.3.2.1 General

The seismic analysis of the reactor coolant system components was performed using normal mode theory in conjunction with time history and response spectrum techniques, as appropriate.

Time history techniques were employed in the analysis of the reactor vessel, the two steam generators, the four reactor coolant pumps and the interconnecting reactor coolant piping. In the analysis of these components, a single composite mathematical model, which included integral representations of each of the components and connecting piping, was employed to account for the
interacting effects of dynamic coupling. The analysis of these dynamically coupled multi-supported components utilized different time dependent input excitations applied simultaneously to each support.

The analyses of the pressurizer and the surge line piping employed separate, uncoupled, mathematical models and utilized response spectrum techniques.

The input data, time histories and response spectra, applied in the analyses were provided by the analysis of the reactor building and internal support structure described in Section 3.7.2.

In all cases except the surge line piping, a damping factor of 1 percent of critical damping was used for each mode. In the analysis of the surge line piping, a damping factor of 0.5 percent of critical damping was used for each mode.

3.7.3.2.2 Mathematical Models

In the descriptions of the mathematical models which follow, the spatial orientations are defined by the set of orthogonal axies where Y is in the vertical direction, and X and Z are in the horizontal plane, in the directions indicated on the appropriate figure. The mathematical representation of the section properties of the structural elements employs a 12 x 12 stiffness matrix for the three dimensional space frame models, and employs a 6 x 6 stiffness matrix for the two dimensional plane frame model. Elbows in piping runs include the in-plane/out-of-plane bending flexibility factors as specified in the USAS B31.7 piping code.

a) Reactor Coolant System - Coupled Components

A schematic diagram of the composite mathematical model used in the analyses of the dynamically coupled components of the reactor coolant system is presented in Figure 3.7-25. This model includes 19 mass points with a total of 47 dynamic degrees of freedom. The mass points and corresponding dynamic degrees of freedom are distributed to provide appropriate representations of the dynamic characteristics of the components, as follows: the reactor vessel, with internals, is represented by 5 mass points with a total of 13 dynamic degrees of freedom; each of the two steam generators are represented by 3 mass points with a total of 7 dynamic degrees of freedom; and each of the four
reactor coolant pumps are represented by 2 mass points with a total of 5 dynamic degrees of freedom. The relatively small mass of the interconnecting reactor coolant piping is lumped proportionately with the masses of the adjoining components.

This mathematical model provides a complete three dimensional representation of the dynamic response of the coupled components to seismic excitations in both the horizontal and vertical directions. The mass is distributed at the selected mass points and corresponding translational degrees of freedom are retained to include rotary inertial effects of the components. The total mass of the entire coupled system is dynamically active in each of the three coordinate directions.

In addition to the model described above, a second model of the coupled components was formulated to incorporate a more detailed representation of the reactor vessel assembly. With the exception of the representation of the reactor vessel assembly, the second model is identical to that shown in Figure 3.7-25. A schematic diagram of the representation of the reactor vessel assembly incorporated into the second model is presented in Figure 3.7-26. This more detailed representation consists of 15 mass points with a total of 33 dynamic degrees of freedom and includes a 10 mass point, 22 dynamic degrees of freedom representation of the reactor vessel internals.

The representation of the reactor vessel internals was formulated in conjunction with the analysis of the reactor vessel internals discussed in Section 3.7.3.3 and was designed to simulate the dynamic characteristics of the models used in that analysis. The second model was used to generate time histories of absolute accelerations at the reactor vessel flange used as forcing functions in the analysis of the reactor vessel internals.

b) Pressurizer

The mathematical model employed in the analysis of the original pressurizer is shown schematically in Figure 3.7-28a. This lumped parameter, planer model provides a multi-mass representation of the axially symmetric pressurizer and includes 5 mass points with a total of 6 dynamic degrees of freedom.

The mathematical model employed in the analysis of the replacement pressurizer is shown schematically in Figure 3.7-28b. This distributed mass, 3-D model provides a representation of the axially symmetric replacement pressurizer, which is fixed at the base. The replacement pressurizer structural model incorporates 3 representative heaters and their support system. The model includes 26 separate node points with a total of 150 degrees of freedom.

c) Surge Line

The lumped parameter, multi-mass mathematical model employed in the analysis of the surge line is shown schematically in Figure 3.7-27. The surge line is modeled as a three dimensional piping run with end points anchored at the attachments to the pressurizer and the reactor vessel outlet piping. In the definition of the mathematical model, 9 mass points with a total of 27 dynamic degrees of freedom were selected to provide a complete three dimensional representation of the dynamic response of the surge line. All supports and restraints defined for the surge line assembly are included in the mathematical model. The total mass of the surge line is dynamically active in each of the three coordinate directions.

3.7.3.2.3 Calculations

a) General

As applied in the analysis, the simultaneous equations of motion for linear structural systems with viscous damping can be written, Reference 4:

$$M\ddot{X} + C\dot{X} + KX = M\ddot{Y} - K_{ms}X_s$$

where:

M = diagonal matrix of lumped masses.

C = square symmetric damping matrix.

K = square symmetric stiffness matrix which defines the mass point force-displacement relationship.

$\ddot{Y}$ = column matrix with elements equal to the absolute acceleration of the datum support in the coordinate direction of the related dynamic degree of freedom of the structural system.

K_{ms} = rectangular matrix of stiffness coefficients which defines the mass point force, non-datum support displacement relationship.

X_s = column matrix of displacements relative to the datum at non-datum supports.

X = column matrix of mass point displacements relative to the datum.

$\dot{X}$ = column matrix of mass point velocities relative to the datum.

$\ddot{X}$ = column matrix of mass point accelerations relative to the datum.

In this form, the equations define the dynamic response of a multimass structural system subjected to time-dependent support motion. In the analysis of systems with multiple supports, such as the coupled components of the reactor coolant system, the equations provide for different time-dependent input motions at each of the supports. In this case, one of the supports of the system is designated the reference, or datum, from which the motions of all other points of the structural system are measured. The reactor vessel support was designated as the datum in the analyses of the coupled components of the reactor coolant system.

Normal mode theory, as described in References 4 and 5, was employed to reduce the equations of motion to a system of independent equations in terms of the normal modes for the time-history and spectrum analyses of the reactor coolant system components. In the analyses, the dynamic response of the components was determined for seismic input excitations in each of the three orthogonal global coordinate directions: X (horizontal), Y (vertical) and Z (horizontal). The dynamic responses to vertical seismic excitation were found for both the case of initial support displacement upward and the case of initial support displacement downward. These responses were combined to determine the most severe combinations produced by the effects of seismic excitations in each of the horizontal directions applied simultaneously with either seismic excitation in the vertical direction.

b) Frequency Analysis

An eigenvalue analysis was performed utilizing the ICES STRUDL II computer code, Reference 6, for the original pressurizer and with BWSPAN, Reference 15, for the replacement pressurizer to calculate the mode shapes and natural frequencies of the composite mathematical models. Modifications to the standard ICES STRUDL II program have been implemented by Combustion Engineering to include a double precision Jacobi diagonalization procedure in the eigenvalue analysis and to provide appropriate influence coefficients and stiffness matrices for use in the response and reaction calculations.

The natural frequencies and dominant degrees of freedom calculated are shown in Table 3.7-22 for all modes used in the analysis of the reactor coolant system surge line and the original pressurizer.

c) Mass Point Response Analysis

The time-history mass point responses to seismic excitation were computed using TMCALC; a C-E code. This code performs a numerical integration of the equations of motion for singly or multiply supported dynamic systems utilizing normal mode theory, Reference 5, and Newmark's BetaMethod with Beta equal to 1/6, Reference 7. For the multiply supported systems, the separate time-histories of each support were imposed on the system simultaneously. The results are time-history responses of the mass points. The analysis of the reactor coolant system utilized modal data for all frequencies through 50 cps.

The mass point responses resulting from the spectrum analysis were found utilizing SHAKE, a C-E computer code. This code performs a normal mode response spectrum analysis resulting in the modal inertial loads at each mass point. The mass point responses of the pressurizer were found using the response spectrum for the pressurizer support. The mass point responses of the surge line were found using an envelope of the support spectra of the interconnected components.

d) Seismic Reaction Analysis

The dynamically induced loads at all system design points due to the time history support excitations and mass point responses were calculated utilizing FORCE, a C-E computer code. This code performs a complete loads analysis of the deformed structure at each incremental time step by computing internal and external system reactions (forces and moments) by superposition of the reactions due to the mass point displacements and the non-datum support displacements as follows:

$$R(t) = C_m X_m(t) + C_s X_s(t)$$

where:

$R(t)$ = the matrix of all components of the reactions at the system design points

C_m = the matrix of mass point displacement influence coefficients

$X_m(t)$ = the column matrix of time story mass point displacements relative to the datum at each time step

C_s = the matrix of support displacement influence coefficients

$X_s(t)$ = the column matrix of time history support displacements relative to the datum at non-datum supports at each time step

The support and mass point displacements due to horizontal and vertical seismic are added algebraically at each time step. The maximum component forces of each reaction for the entire time domain, and its associated time of occurrence, are selected.

The square root of the sum of the squares method is the procedure normally used to combine the modal responses when the modal analysis response spectrum method of analysis is employed. The procedure is modified only in two cases:

a) In the analysis of simple systems where three or less dynamic degrees-of-freedom are involved, the modal responses are combined by the summation of the absolute values method.

b) In the analysis of complex systems where closely spaced modal frequencies are encountered, the responses of the closely spaced modes are combined by the summation of the absolute values method and, in turn, combined with the responses of the remaining significant modes by the square root of the sum of the squares method. Modal frequencies are considered closely spaced when their difference is less than $\pm$ 10 percent of the lower frequency.

The maximum reactions for the original pressurizer, replacement pressurizer and surge line resulting from the response, spectrum analysis were found by applying the modal inertial loads for each mode, to the structural model. This was done using the ICES STRUDL II computer code for the original pressurizer and surge line and using the BWSPAN computer code for the replacement pressurizer. The design point reactions due to each modal loading were conservatively combined by summing the absolute values of the modal reactions. The surge line analysis included consideration of the relative end displacements. The reactions found by statically imposing the maximum relative displacements of the two ends of the surge line were conservatively included by absolute summation with the inertial response from the spectrum analysis.

3.7.3.2.4 Results

The demonstration of design adequacy is made by a comparison of the loads specified in the component equipment specifications with those determined through dynamic seismic analysis. The margins between the specified loads and the loads determined by dynamic analysis demonstrate margin between the stresses that would result from seismic loading and the stresses that have been proven to be acceptable through the design stress reports.

The reactions (forces and moments) at all design points in the system, obtained from the dynamic seismic analysis, were compared with the seismic loads in each component design specification. The results of this comparison are summarized in Table 3.7-23 for the points of maximum calculated load.

The maximum seismic loads calculated by the time history techniques are the result of a search and comparison over the entire time domain of each individual component of load due to the simultaneous application of the horizontal and either vertical excitation. The maximum calculated components of load shown in Table 3.7-23 for each design location, in general, occur neither at the same time nor for the same combination of horizontal and vertical excitation, and therefore result in a conservative case.

All original calculated maximum loads of Table 3.7-23 have been conservatively increased and evaluated to account for the increase in mass and center of gravity of the replacement steam generator. The increased loads, RSG Maximum, remained below the specified design loads.

The maximum seismic loads calculated by the response spectrum techniques are the result of combining the modal reactions due to the horizontal and the vertical excitation on an absolute sum-basis.

The results shown are for the Operational Basis Earthquake. For determination of results due to the Design Basis Earthquake, both the calculated results and specification values are multiplied by a factor of 2.0.

It is concluded that the seismic loadings specified for the design of the reactor coolant system components and supports are adequate for the Operational Basis Earthquake and the Design Basis Earthquake conditions. All seismic loads calculated by the dynamic seismic analysis are less than the corresponding loads in the component design specification.

3.7.3.3 Seismic Analysis - Reactor Internals and Core

3.7.3.3.1 Introduction

Dynamic analyses of the reactor vessel internals and core were conducted to determine their response to horizontal and vertical seismic excitation and to verify the adequacy of their seismic design. All reactor internals are classified as Category I for seismic design purposes. The dynamic seismic analysis of the internals and core included the use of modal analyses techniques utilizing both response spectra and time-history accelerograms for linear conditions, and step-by-step integration of the equations of motion for nonlinear impact conditions such as exists when the gaps between components close. These analyses were conducted in conjunction with the analyses of the reactor coolant system as discussed in 3.7.3.2. The following sections provide a description of the mathematical models and analytical procedures used for the internals and core, and the applicable stress and deformation criteria. Analysis and testing of the control rod drive system under the influence of seismic excitation is discussed in Section 4.2.3.

3.7.3.3.2 Seismic Load, Stress and Deformation Criteria

a) General

The seismic loads on the reactor internals and core are combined with normal operating loads and postulated accident loads for the plant conditions categorized as upset, emergency and faulted.

1) Upset Conditions

During upset conditions, the reactor internals and core are required to perform their function without shutdown. The loads from the OBE are combined with the normal operating loads.

2) Emergency Conditions

Under emergency conditions, some local yielding of the reactor internals and core is allowed when subjected to combined normal operating and DBE loads. A small number of fuel elements may be damaged.

3) Faulted Conditions

The loading combination for these conditions includes the loads resulting from normal operation, DBE and postulated LOCA. Permanent deformation is permitted. Deflections are limited so that the core will be held in place and adequate core cooling is preserved.

b) Stress Limits

The stress values for the reactor internals under the above conditions are not greater than those given in the May 1972, draft of Section III of the ASME Boiler and Pressure Vessel Code, Subsection NG, including Appendix F, "Rules for Evaluation of Faulted Conditions". The stress limits and loading conditions for the internals and core are summarized in Table 3.7-24.

For critical reactor internals components which are subjected to fatigue, the design fatigue curve of Figure I-9.2 of Section III of the ASME Boiler and Pressure Vessel Code and cumulative usage factor of less than one was utilized.

c) Deformation Limits

In addition, to properly perform their functions, the reactor internal structures satisfy the deformation limits listed below.

1) Under design loadings plus operating basis earthquake forces or normal operating loadings plus design basis earthquake forces, deflections are limited so that the control element assemblies (CEA's) can function and adequate core cooling is preserved.

2) Under normal operating loadings plus design basis earthquake forces plus pipe rupture loadings resulting from an equivalent diameter pipe break not exceeding the largest line connected to the main reactor coolant lines, deflections are limited so that the core is held in place, adequate core cooling is preserved, and all control element assemblies can be inserted. Those deflections which would influence CEA movement are limited to less than 80% of the deflections required to prevent CEA insertion.

3) Under normal operating loadings plus design basis earthquake forces plus maximum pipe rupture loadings resulting from the full spectrum of pipe breaks, deflections are limited so that the core will be held in place and adequate core cooling is preserved.

Although CEA insertion is not required for a safe and orderly shutdown for breaks larger than equivalent diameter pipe break not exceeding the largest line connected to the main reactor coolant lines, calculations show that the CEA's will be insertable for large breaks except for a few CEA's located near the vessel outlet nozzle, which is feeding the postulated rupture.

The core deformation limits and functional requirements for seismic conditions are as follows:

1) Operational Basis Earthquake (OBE)

a. The CEA's can be scrammed during the OBE within the allowable scram time.

b. No inspection of fuel or CEA's is required for continued operation.

2) Design Basis Earthquake (DBE)

a. The fuel assemblies must be capable of sustaining a DBE with no loss of function with respect to safety.

b. The CEA's may scram during a DBE, but not necessarily within the prescribed time.

c. After a DBE, tests and inspections may be required before resuming normal operation.

3) Faulted Conditions

Permanent deformation of the fuel assembly is permitted during accidents (including DBE and LOCA) which are not expected to occur.

a. If the equivalent diameter pipe break does not exceed the largest line connected to the main reactor coolant lines, the deformation shall be limited, permitting the CEA's to scram.

b. For major pipe breaks, the fuel shall be held in a coolable array.

3.7.3.3.3 Method of Analysis

a) General

The procedure used in conducting the seismic analysis of the reactor internals consisted basically of three steps. The first step involved the formulation of a mathematical model. The natural frequencies and mode shapes of the model were determined during the second step. The response of the model to the seismic excitation was determined in the third step. In this analysis, the horizontal and vertical components of the seismic excitation were considered separately and the maximum responses added to obtain conservative results.

b) Mathematical Models

Equivalent multi-mass mathematical models were developed to represent the reactor internals and core. The linear mathematical models of the internals were constructed in terms of lumped masses and elastic beam elements. At appropriate locations within the internals and core, points (nodes) were chosen to lump the weights of the structure. A sketch of the internals and core showing the relative node locations for the horizontal model is presented in Figure 3.7-29. Figures 3.7-30 and 3.7-31 show the idealized linear horizontal and vertical models. The criteria for choosing the number and location of mass concentration was to provide for accurate representation of the dynamically significant modes of vibration of each of the internals components. Between the nodes, properties were calculated for moments of inertia, cross-section areas, effective shear areas and lengths. Since the seismic excitation of the internals was input at the vessel/internals interface, only the internals and core are included in the models. Separate horizontal and vertical models of the internals and core were formulated to more efficiently account for structural differences in these directions. Consequently, the horizontal and vertical responses were treated as uncoupled. Since the structural details provide for no vertical load transfer between the upper guide structure and core or core shroud, the vertical response of the upper guide structure is independent of the rest of the internals and core. Therefore, the vertical model was divided into two sub-models. Model A consists of the core support barrel, lower support structure, core shroud and core mass; Model B consists of the upper guide structure.

The salient details of the internals and core models are discussed below:

1) Hydrodynamic Effects

The dynamic analysis of reactor internals presents some special problems due to their immersion in a confined fluid. It has been shown both analytically and experimentally (Reference 1) that immersion of a body in a dense fluid medium lowers its natural frequency and significantly alters its vibratory response as compared to that in air. The effect is more pronounced where the confining boundaries of the fluid are in close proximity to the vibrating body as is the case for the reactor internals. The method of accounting for the effects of a surrounding fluid on a vibrating system has been to ascribe to the system additional or "hydrodynamic mass".

This "hydrodynamic mass" decreases the frequencies of the system, but is not directly involved in the inertia force effects. The hydrodynamic mass of an immersed system is a function of the dimensions of the real mass and the space between the real mass and confining boundary.

Hydrodynamic mass effects for moving cylinders in a water annulus are discussed in References 8 and 9. The results of these references were applied to the internals structures to obtain the total (structural plus hydrodynamic) mass matrix which was then used in the evaluation of the natural frequencies and mode shapes for the model.

2) Fuel Assemblies

For the linear horizontal model, the fuel assemblies are treated as vibrating in unison. The member properties for the beam elements representing the fuel assemblies were derived from the results of experimental tests of the fuel assembly load deflection characteristics and natural frequency (Reference 10).

3) Core Support Barrel Flanges

To obtain accurate lateral and vertical stiffness of the upper and lower flanges finite element analyses of these two regions were performed. As shown in Figure 3.7-32 for the upper flange, these areas were modeled with quadrilateral and triangular ring elements. Asymmetric loads, equivalent to lateral shear loads and bending moments, and symmetric axial loads were applied and the resulting displacements calculated. These results were then used to derive the equivalent member properties for the flanges.

4) CEA Shrouds

For the horizontal model, the CEA shrouds were treated as vibrating in unison and were modeled as guided cantilever beams in parallel. To account for the decreased lateral stiffness of the upper guide structure due to local bending of the fuel alignment plate, a short member with properties approximating the local bending stiffness of the fuel alignment plate was included at the bottom of the CEA shrouds. Since the stiffness of the upper guide structure support plate is large compared to that of the shrouds, the CEA shrouds were assumed to be rigidly connected to the upper guide structure support plate.

5) Thermal Shield (historical)

During the March, 1983 refueling outage, difficulties were encountered during core reload when a fuel assembly would not seat properly on the core support plate. Subsequent inspection determined there was debris of unknown origin on the plate. The fuel was unloaded and the core support barrel was removed to investigate the source of the debris.

A visual examination of the core support barrel/thermal shield assembly disclosed the thermal shield support system to be damaged. A number of thermal shield support pins were fractured and/or missing and damage to the core support barrel was visible.

An evaluation of the thermal shield support system concluded that refurbishment was impractical. Therefore, a decision was made to remove the thermal shield. Analyses performed to evaluate operation of the plant without a thermal shield for its remaining design life indicated that replacement of the thermal shield was not necessary. FPL has reviewed these analyses and determined that the modifications, while slightly increasing the reactor vessel fluence, is acceptable for reactor vessel integrity, as documented in FPL Letter L-84-29 and NRC letter of March 14, 1984. Refer to Section 4.2.2.2 for details of thermal shield removal and reanalysis of the reactor vessel and internals.

6) Upper Guide Structure Support Plate
& Lower Support Structure Grid Beams

These grid beam structures were modeled as plane grids. Displacements due to vertical (out of plane) loads applied at the beam junctions are calculated through the use of the STRUDL computer code (Reference 6). Average stiffness values based on these results yielded equivalent member cross-section areas for the vertical model.

c) Natural Frequencies and Mode Shapes

The mass and beam element properties of the models were utilized in STAR, a computer program from the MRI/STARDYNE Analysis System programs (Reference 11) to obtain the natural frequencies and mode shapes. This system utilizes the "stiffness matrix" method of structural analysis. The natural frequencies and mode shapes are extracted from the system of equation.

$$\left[\underline{K} - W_n^2 \underline{M}\right] \underline{\Phi}_n = 0$$

where:

$\underline{K}$ = Model stiffness matrix

W_n = Natural circular frequency for the n^{th} mode

$\underline{M}$ = Model mass matrix

$\underline{\Phi}$ = Normal mode shape matrix for n^{th} mode

The mass matrix, $\underline{M}$, includes the hydrodynamic and structural masses.

The natural frequencies and mode shapes calculated for the first 3 modes for the horizontal model are presented in Figures 3.7-33 through 3.7-35. The natural frequencies calculated for the vertical model are presented in Table 3.7-25. The model data shown is typical and is presented for illustrative purposes. The effect of additional higher modes was included in the response analyses.

d) Response Calculations

1) Horizontal Direction

The time history analysis technique was utilized to obtain the response of the internals for the horizontal seismic excitation. The horizontal excitation was specified as the acceleration time history of the reactor vessel flange, resulting from the OBE. The flange excitation resulting from the DBE was conservatively specified as 2 times that for the OBE. This excitation was determined in the analysis of the reactor coolant system as discussed in Section 3.7.3.2.

The time history response analysis was performed utilizing the MRI STARDYNE System/DYNRE 1 Computer Program. This program utilizes the "Normal Mode Method" to obtain time history response of linear elastic structure. Details of the program and the "Normal Mode Method" are presented in References 5, 11 and 12.

Input to DYNRE 1 consisted of the modal data as determined in Section 3.7.3.3.3c modal damping factors, and the forcing function time history. This analysis used the modal data for all modas with frequencies below 100 cps. This included the first 14 modes. Contributions from higher modes are negligible.

The modal damping factors were obtained by the method of "Mass Mode Weighing" which gives:

$$\beta_n = \frac{\Sigma M_i \left| \phi_{in} \right| \beta_i}{\Sigma M_i \left| \phi_{in} \right|}$$

where: β_n = Modal damping factor

M_i = Structural mass of mass node i

$|\phi_{in}|$ = Absolute value of the mode shape as mass mode i

β_i = Damping associated with mass point i

The damping factor assigned to the nodes representing the fuel assemblies was 5 percent. This is a conservative value derived from experimental results (Reference 6). A value of 1 percent was used for the other nodes.

The output from the DYNRE 1 code consists of the nodal displacement, velocity, and acceleration time history relative to the base. The member bending moments and shears were obtained from the STAR code (Reference 11) and were derived from the DYNRE 1 nodal displacement vectors at the time of peak response.

2) Vertical Direction

The response of the reactor internals to the vertical excitation was obtained by the response spectrum technique. Because of the high natural frequencies and resulting low levels of response for the vertical direction, the more conservative spectrum response analysis results were used instead of time history results. The response spectrum utilized was derived from the vertical acceleration time history at the reactor vessel flange. The spectrum curve is presented in Figure 3.7-36.

An acceleration level corresponding to the natural frequency of each mode was selected from the spectrum curve. The response spectrum technique used these acceleration values to determine the inertia forces, accelerations, and displacements of each mode. The results for each mode were conservatively combined on the basis of absolute values. For the vertical models, the first seven modes were included in the results. Contributions from modes above the first were small.

3) Nonlinear Impact Analysis

The linear horizontal analysis indicated that the fuel assembly spacer grids contact the core shroud. Analyses were performed to determine the magnitude of the impact loads on the spacer grids. The nonlinear response and forces were determined using the "SHOCK" computer code (Reference 13). This computer code provides the numerical solution to transient dynamic problems which have been modeled as lumped-parameter systems by step-by-step integration of the equation of motion. The model of the fuel assemblies contained concentrated masses and nonlinear springs located at the spacer grids. The nonlinear springs represented the springs located at the spacer grids. The nonlinear springs represented the gap between spacer grids and the core shroud and the spacer grid stiffness, which was obtained from tests on production grids. The results from the linear time history analysis just before impact were used as initial conditions for the nonlinear model.

3.7.3.3.4 Results

Combined results for the horizontal and vertical dynamic seismic analyses are presented in Table 3.7-26 in terms of stresses at critical locations in the reactor internals for the DBE. Table 3.7-26 also lists the seismic stresses which result from the application of the design loads specified for the DBE. A comparison shows the results of the dynamic analysis to be less severe.

For the OBE, loads on the fuel assembly spacer grids as determined from the nonlinear analysis were less than the experimentally determined elastic limit load. For the DBE loads on the spacer grids were well under the load required to distort the guide tubes and prevent CEA insertion.

The input to the reactor internals seismic analysis is the reactor coolant system seismic analysis, which was not revised for EPU conditions. Therefore, the reactor internals are not re-analyzed for seismic events, as the pre-EPU analysis remains applicable for EPU.

3.7.3.4 Method of Analysis - Other Seismic Class I Systems

3.7.3.4.1 General

All seismic Class I piping systems are analyzed to assure non-resonance with supporting structures in the following manner:

a) Piping 2½ inches nominal size or larger with design temperature above 275 F is analyzed by the multi-modal method as described in 3.7.3.4.2.

b) Piping 2½ inches nominal size or larger with design temperature up to 275 F is analyzed by the simplified method as described in 3.7.3.4.3.

c) Piping from l¼ to 2 inches nominal size with design temperature above 275 F is analyzed by the simplified method as described in 3.7.3.4.3.

d) Piping from l¼ to 2 inches nominal size with design temperature up to 275 F and piping from ½ to 1 inch nominal size for all design temperatures is analyzed and restrained to the extent of specifying the appropriate maximum span length between supports such that the component period is less than 70 percent of the structure second mode period.

The larger size piping systems with design temperatures above 275 F are not analyzed by the simplified method since this would normally result in the placing of a greater number of restraints than would result from analysis by the multi-modal method. Since these lines must be allowed greater thermal movement, the simplified method, though conservative for seismic considerations, often becomes too restrictive for thermal considerations.

For high temperature lines that are analyzed by the multi-modal method, a flexibility analysis is first made and if stresses are within acceptable limits, restraints for seismic protection are located at points of negligible thermal movements. The system is then analyzed seismically and reanalyzed thermally. If not acceptable, restraints are relocated and the system is reanalyzed. This procedure is repeated as often as necessary to achieve proper design, and only as a last resort are hydraulic snubbers specified. For low temperature lines, restraints are tentatively located along the pipe run and a thermal evaluation is made, as described in Section 3.7.3.4.3. If required, restraints are relocated and the procedure is repeated as necessary.

The time-history earthquake records at ground level as described in Section 3.7.1 were used to calculate time-history structural responses at the system supports. This time-history of response was then used to compute the floor response spectra shown on Figures 3.7-12 through

3.7-24. These spectra are then utilized in the analysis of the mechanical systems.

For systems which are analyzed by the multi-modal method, if the first mode period of the piping is 70 percent or less of the second mode period of the structures, the full multi-mode response analysis is not performed. The static analysis is made directly, using an acceleration value of 1.5 times the maximum value of the floor response spectrum in the period range equal to or less than the period of the piping.

For systems analyzed by the simplified method, the preset value for the maximum allowable period is 70 percent of the second mode period of the structures. The acceleration values used for the design of restraints are 1.5 times the maximum value of the floor response spectrum in the period range from zero to the 70 percent value.

3.7.3.4.2 Multi-Modal Seismic Analysis Method

The method of dynamic analysis by the multi-modal method is described as follows:

a) Basic Assumptions

1) The system is linearly elastic.

2) Masses are lumped at discreet points and are connected by weightless elastic members.

3) Each mass point has six degrees of freedom except for points indicated as restrained in a given direction.

4) The system is anchored at two or more positions and these anchor points are fixed for the determination of natural frequencies and mode shapes.

5) Dynamic loadings in the three coordinate directions are determined separately and combined on the basis of excitation occurring in the vertical and one horizontal direction at the same time.

6) The mass polar moment of inertia, i.e., the mass component involved in rotation, is negligible.

7) Damping is viscous and assumed constant for all modes.

8) Increased flexibility due to pipe bends is included in the analysis.

b) Equations of Motion

The stiffness matrix method of natural mode analysis is employed to determine natural periods of vibration and the associated mode shapes.

The equations of motion for the piping system may be written as

$$[M]\{\ddot{\Delta}\} + [K]\{\Delta\} = \{F\} \tag{1}$$

$[M]$ = Diagonal matrix of lumped masses, the rows and columns of which are arranged to correspond to the components of the stiffness matrix. The masses effective in the three coordinate directions are taken to be equal to the total mass assumed lumped at the point under study.

$[K]$ = Square, symmetric matrix of stiffness coefficients including the effects of axial deformation, bending and torsional shear in the three coordinate directions.

$\{\Delta\}$ = Column matrix of displacement.

$\{\ddot{\Delta}\}$ = Column matrix of acceleration.

$\{F\}$ = Column matrix of external loads.

The stiffness matrix $[K]$ is assembled as follows:

Each pipe section has the properties:

E = Modulus of Elasticity

μ = Poisson's Ratio

I = Moment of Inertia

A = Cross-sectional Area

L = Length

From these properties the characteristics of the section are computed:

$$G = \frac{E}{2\ (1 + \mu)};\ GJ = \frac{EI}{1 + \mu};\ \alpha = \beta = \frac{2\,(1 + \mu)}{AE}$$

$$\in\ = \eta = \frac{1}{EI};\ Y = \frac{1}{AE};\ \lambda = \frac{1}{GJ} = \frac{1 + \mu}{EI}$$

The end flexibility of the section is contained in the 6 x 6 matrix ϕ:

$$\phi = \begin{Bmatrix} \alpha L + \dfrac{\in L^3}{3} & 0 & 0 & 0 & \dfrac{\in L^2}{2} & 0 \\ 0 & \beta L + \dfrac{\eta L^3}{3} & 0 & -\dfrac{\eta L^2}{2} & 0 & 0 \\ 0 & 0 & \gamma L & 0 & 0 & 0 \\ 0 & -\dfrac{\eta L^2}{2} & 0 & \eta L & 0 & 0 \\ \dfrac{\in L^2}{2} & 0 & 0 & 0 & \in L & 0 \\ 0 & 0 & 0 & 0 & 0 & \lambda L \end{Bmatrix}$$

A rotation matrix [R] is established to bring the pipe section into the general coordinate system. This matrix is based on the orientation and location of the section in the overall system.

The flexibility in the generalized coordinate system is:

$[\phi_G] = [R]\,[\phi]\,[R]^T$

The flexibilities $[\phi_G]$ are accumulated for each mass point and the stiffness coefficients are computed as

$[K]_A = [\phi_G]^{-1}$

and assembled into the overall stiffness matrix [K].

For the determination of natural frequencies and mode shapes, equation one is solved by first setting the external loads {F} equal to zero and the displacement vector $\{\Delta\} = \{\delta\} \sin \omega t$.

Then:

$\{\ddot{\Delta}\} = -\{\delta\}\,\omega^2 \sin \omega t.$

Equation (1) becomes:

$$[K]\,\{\delta\} = \omega^2\,[M]\,\{\delta\} \qquad (2)$$

This characteristic eigenvalue equation is solved by iterative techniques to determine the natural frequencies and mode shape vectors {δ} of the system.

This generalized procedure permits the analysis of multiple fixed branched and looped systems with multiple lumped masses as well as simple single branch systems.

c) Modal Analysis

The response of each mode of vibration considered was computed for a unidirectional earthquake disturbance in the X or Z direction and the vertical Y direction simultaneously as:

$$R_{nd} = \sum_{i=1}^{n} M_i \delta_{idn}$$

where:

M_n = Effective mass = $\sum_{d=1}^{d_3} \sum_{i=1}^{n} M_i \delta^2_{idn}$; n = Mode number;

d = Direction X, Y, Z; N = Total number of lumped masses

S_{and} = Floor response spectral acceleration in the d direction for the nth mode;

δ_{idn} = Shape factor (i^{th} component for the n^{th} mode shape for direction d)

The disturbance factor for the earthquake in one horizontal coordinate direction and the vertical direction is defined as:

$$D_n = \left| R_{nd_1} \right| S_{and_1} + \left| R_{nd_2} \right| S_{and_2}$$

where d_1 and d_2 indicate horizontal and vertical directions respectively.

The model inertia forces for each mode of vibration are then computed as:

$$F_{idn} = \frac{M_i \delta_{idn} D_n}{M_n}$$

d) Stress and Displacement Analysis

The modal inertia forces F_{idn} are utilized as response loads in a static analysis to generate modal internal forces F^*_{idn}, moments M^*_{idn} and displacement Δ_{idn}, the final stresses resulting from the earthquake disturbance in one horizontal coordinate direction and the vertical direction are computed as the maximum resulting from combining the modal stress by the square root of the sum of squares method. The final inertia of shear forces, moment and displacement to be used for design are determined by combining the results of the modes considered on the same basis.

i.e.,

$$F^*_{id} = \left(\sum_n F^*_{idn}{}^2 \right)^{1/2}$$

$$M^*_{id} = \left(\sum_n M^*_{idn}{}^2 \right)^{1/2}$$

$$\Delta_{id} = \left(\sum_n \Delta_{idn}{}^2 \right)^{1/2}$$

$$\sigma_i = \left(\sum_n \quad {}_{in}{}^2 \right)^{1/2}$$

$$\sigma_{in} = \left(\sum_{d=d}^{d_3} \frac{M^*_{idn}{}^2}{Z} \right)^{1/2}$$

The intensification factor will be applied to bending moment only.

The computer program used for the static analysis utilizes the same stiffness matrix method as that described for the dynamic analysis. The program determines forces, moments and deflections in the three coordinate directions and the stresses at selected points in the piping system.

Piping systems that pass between structures are analyzed by the piping analysis program for stresses created by the relative displacement of anchor points that are located in different structures.

3.7.3.4.3 Simplified Seismic Analysis Method

The simplified method of analysis consists of locating restraints such that the period of the first mode of vibration of the piping system will not exceed the preset value of 70 percent of the second mode period of the structure. This method involves the use of appropriate and comprehensive charts and tabulations that include correction factors for the effects of concentrated loads, branch connections, changes in pipe size, changes of direction, offsets and various combinations of these effects in each of the three coordinate directions to assure that it is adequately restrained in all directions. An additional analysis is performed to evaluate the thermal effects of the restraints system. This is done by means of charts that define the minimum distance required for placing restraints adjacent to any expanding leg in order to stay within allowable stress limits.

The floor response spectra for the reactor building and reactor auxiliary building were generated by using the synthetic time-history ground motion as described in Section 3.7.1 with the soils moduli of elasticity as determined in Section 2.5.4. The reactor building and reactor auxiliary building response spectra for the operating basis earthquake (OBE)(0.05 g horizontal ground acceleration) with 0.5 percent damping are shown on Figures 3.7-12 through 3.7-24. The response spectra for the design basis earthquake (DBE) (0.10 g horizontal ground acceleration) is obtained by doubling the acceleration values for the (OBE) response spectra.

In the floor response spectra each curve contains two significant peaks which are associated with the first mode and second mode of the building.

A study of the floor response spectra curves shows that there is ample justification for using 70 percent of the second mode period of the structures as a design basis for the piping. Typically, the curves depicting the response of the structures peak very sharply for the first and second mode, and there is a noted absence of any other significant peaks for the higher modes. It is then obvious that if the primary mode of piping systems are kept within the 70 percent factor, there is no need for further stringent response analysis.

Table 3.7-27 is a summary of the pertinent data from the floor response spectra. It shows that for any direction of earthquake excitation, the values to be used for the design period of the piping is based on the minimum period of the structures and that the design accelerations are based on the maximum structural accelerations within the design periods of the piping.

If the primary period of vibration of a piping system is kept below 70 percent of the horizontal second mode periods (0.20 for reactor building and 0.15 for reactor auxiliary building) resonance with the supporting structure at the peak accelerations is avoided. It is then sufficient to design for the maximum support accelerations between 0 to 70 percent of the second mode period of the structure with a conservative combined participation factor applied for the higher modes. A factor of 1.5 was chosen based on past experience with participation factors for the higher modes which shows such a value to be conservative.

3.7.3.4.4 Comparison of Typical Results of Multi-Modal Method and Simplified Method

A number of sample problems were analyzed in order to compare the results of the simplified method with the multi-modal method on the same piping systems. In the first group of test problems, restraints were located on the typical piping systems shown in Figure 3.7-37 by using the criterion of the simplified method for not exceeding a preset period limit. The periods were then checked by computer analysis, and were found to be well within the limits, as shown on Table 3.7-28.

The table also includes the results of a second set of test problems that were made to show the conservatism in using 1.5 as the participation factor for higher modes. The typical piping systems shown in Figure 3.7-38 which have primary mode period of approximately 0.20 sec., were conservatively analyzed first by the full response multi-modal method for various modes with an acceleration value of 1.0 g for each mode, and then by the direct static method with an acceleration value of 1.5 g. Results show that for these periods of vibration, the 1.5 participation factor is extremely conservative.

3.7.3.5 Torsional Effects of Valves

Torsional effects of motor and air operated valves and other eccentric masses are included in the analysis of reactor coolant pressure boundary piping by taking into account the mass and its eccentricity in the mathematical model.

For piping outside the reactor coolant pressure boundary, the weight of valves and operators is lumped at the center line of the pipe in the mathematical model except for lines 2 inches or less in diameter.

When analysis using the lumped procedure indicates stresses close to allowable code values in the vicinity of the valve, the stress analysis is performed considering the eccentricity of the masses.

3.7.3.6 Differential Movement of Piping Supports

Differential movement between floors of a building is neglected, due to its minimal effect, in the seismic analysis of components. However, differential movement between buildings as well as differential settlement of foundations, if not on the same mat, is considered.

For seismic Class I piping interconnecting structures or structures and the ground, the relative displacements due to the design basis earthquake are determined from the dynamic analysis of the structures. The piping is designed for the Maximum axial and lateral displacements resulting from this analysis.

For piping between adjacent structures, if there is not sufficient flexibility in the piping system, expansion loops are used to absorb the differential axial displacement. For lateral displacement, pipe restraints are provided with sufficient opening to prevent exceeding the stress limits on the piping. The section of piping between the anchor at a structure and the first restraint is treated as a deflected cantilever beam in order to estimate the maximum stresses due to the differential displacement. The bending stress is calculated as follows:

$$\sigma\ bending = \frac{3Ec\Delta}{L^2}$$

where:

c = (pipe OD)/2

E = Modulus of elasticity (psi)

Δ = Relative lateral displacement

L = Distance

For seismic Class I piping between a structure and the ground, the piping is provided with flexible joints to permit differential motion between the building and ground, or the piping is run underground for some distance in a culvert that allows sufficient displacement of the piping without creating an overstressed condition.

Buried seismic Class I piping is assumed to be distorted in the same fashion as the earth, hence, assumes a sinusoidal wave shape. The wave length and the maximum displacement is calculated and the bending moments and stress are determined.

3.7.3.7 Interaction with Non-Class I Systems

Where seismic Class I piping systems are Connected to non-Class I systems, an anchor is located at the interface of the Class I and non-Class I piping where practical. If this is not possible, the non-Class I piping is treated as Class I piping up to the first anchor or combination of restraints which limit the seismic effects of the non-Class I piping on the Class I piping and anchor. The loads due to the non-Class I piping past the anchor are added to the anchor loads obtained from the Class I piping analysis.

3.7.3.8 Field Location of Seismic Restraints and Supports

Seismic restraints, supports and snubbers are shown on the stress analysis isometrics and these are used by the fabricator to prepare detail and location drawings. The installed restraints, supports and snubbers are inspected to assure proper location and function.

Seismic Class I field run piping is analyzed and restrained by specifying the maximum span length between supports such that the component period is less than 70 percent of the structure second mode period. Installation of seismic restraints on field run piping is checked and inspected in the same manner as for piping which is not field run.

3.7.3.9 Reactor Building Crane Restraints

The reactor building polar crane is provided with bold down lugs as shown in Figure 3.7-39 which will prevent the crane from being dislodged from its rails in the event of a DBE. The reactor building one ton telescoping jib crane is restrained and seismically qualified in the restrained position to assure that it will not become dislodged in the event of a DBE.

3.7.3.10 Additional Computer codes used in Piping System Stress Analysis and Support Design

PIPESTRESS:
PIPESTRESS program performs linear elastic analysis of three dimensional piping systems subject to a variety of loading conditions. Piping systems may be investigated for compliance with piping codes and with other constraints on system response.

PC-PREPS:
This computer code uses linear elastic theory to analyze and qualify structural members, welds, local tube steel stresses, anchor bolts, surface mounted base plates in accordance with AISC, AWS and/or ASME criteria.

PITRUST-PC:
This computer code calculates local stress intensity at the junction of two cylindrical vessels. The method and theory of calculating stresses follow that promulgated by Welding Research Council (WRC) bulletin No. 107. This computer code is capable of complying with requirements of ASME Boiler and Pressure Vessel Code, Section III and ANSI B31.1 piping codes.

RELAP5:
The RELAP5 computer code is a PC based QA Category 1 light water reactor transient analysis code developed at the Idaho National Engineering Laboratory (INEL) for the NRC. It is a highly generic code that, in addition to calculating the behavior of a reactor coolant system (RCS) during a transient, can be used for simulation of a wide variety of hydraulic and thermal transients in both nuclear and non-nuclear systems involving mixtures of steam, water, non-condensable and solute. This computer code was used to determine forcing functions for the pressurizer power operated relief valve (PORV) opening event.

STEHAM-PC:
The STEHAM-PC computer program is a generalized fluid transient analysis code that is used to perform steady-state and transient analyses of steam filled flow network. This code was used to determine forcing functions for a main steam isolation valve closure event and a turbine stop valve closure fluid transient event.

WATHAM-PC:
The WATHAM-PC computer program is a generalized fluid transient analysis code that is used to perform transient analyses of water filled flow network due to pump start-up, pump trip and valve opening and closing. This code was used to determine forcing functions for the feedwater regulating valve and isolation valve closure, and feedwater pump trip events.

3.7.4 SEISMIC INSTRUMENTATION PROGRAM

3.7.4.1 Compliance with Regulatory Guide 1.12

The original seismic monitoring system was evaluated against the requirements of Regulatory Guide 1.12, Revision 01, "Instrumentation For Earthquake," and was found to be acceptable. The seismic monitoring system was upgraded in 2004. Where possible the replacement seismic monitoring system was designed to minimize the need for new interfacing signal cable, and consequently many of the original system design features (e.g., monitor locations, power sources, etc.) were retained. The following constitutes an item-by-item evaluation of the current seismic monitoring system design vs. the requirements of Regulatory Guide 1.12, Revision 02. All system design characteristics meet the requirements of Regulatory Guide (RG) 1.12, or have not been changed from the original system design and are acceptable on that basis.

1. Seismic Instrumentation Type and Location:
The new seismic monitoring system design is based on Kinemetrics Etna monitors, which are solid state digital instruments. Software will be maintained on a portable computer to facilitate download and analysis of seismic event response data from the monitors within four hours. The locations of the five original triaxial accelerometers (free field, containment foundation, containment upper elevation, RAB foundation & RAB upper elevation) were retained in the new seismic monitoring system design. These locations are generally consistent with the RG requirements, and were found to be acceptable as documented in the Unit-1 SER.

2. Instrumentation At Multi-Unit Sites:
Based on structural similarity between Units 1 & 2, a common seismic monitoring system is acceptable. System alarms are annunciated in the main control room of both Units.

3. Seismic Instrumentation Operability:
Operability requirements for the seismic monitoring system are maintained in UFSAR sections 13.8 and 13.7 for Units 1 & 2 respectively.

4.1 Instrumentation Characteristics:
The triaxial accelerometer section of each Etna monitor includes a calibration coil that can be actuated via the communication link to perform a complete functional test of the monitor. RG 1.12 requires minimum pre and post event data recording times of 3 and 5 seconds respectively. Pre and post-event data recording times of 10 seconds each are used for the new Etna monitors. RG 1.12 requires a minimum data recording capability of 25 minutes. With the standard 32MB flash memory card, the Etna can record approximately 256 minutes of data, based on a sampling rate of 200 SPS for each of the three triaxial channels. The minimum standard Etna battery capacity is 6.5 AH. Based on the specified normal Etna current drain of 185 ma, the battery can supply the monitor for 35 hours without the AC input.

4.2 Acceleration Sensors and Recorder:
Since a single instrument performs both functions, the RG 1.12 requirements for the sensors and recorder will be evaluated together. The dynamic range of the Etna is user selectable. A range of $\pm 2g$ will be used, which meets the requirements of RG 1.12. The frequency range or bandwidth of the Etna is dependent upon the selected sample rate. Based on an expected sample rate of 200 SPS, the bandwidth is 0 to 80 hz, which meets the RG 1.12 requirements.

4.3 Seismic Trigger:
A common trigger/de-trigger setpoint of 0.01g (any triaxial channel) will be used for all Etna monitors. This is consistent with the original system design and with the Reg Guide 1.12 requirements.

5. Instrumentation Installation:
The seismic monitoring system upgrade did not alter the locations or mounting details of the seismic monitors. These details are consistent with the Kinemetrics installation instructions, and with the Reg Guide 1.12 requirements.

6. Instrumentation Actuation:
The Etna monitors will be configured to trigger at a setpoint of 0.01g in any triaxial direction. This is consistent with the existing system design and with the Reg Guide 1.12 requirements.

7. Remote Indication:
The seismic monitoring system upgrade did not alter the audible alarm features of the original seismic monitoring system. Annunciation is provided in both main control rooms if the containment building foundation monitor (SMR-42-1) triggers.

8. Maintenance:
Surveillance requirements for the seismic monitoring equipment are contained in Sections 13.8 and 13.7 of the Unit 1 & 2 UFSAR respectively.

g) Position 7: The acceleration data recorded at EL 23 ft in the reactor containment is available in the control room via a communication link. Refer to Section 3.7.4.3.

h) Position 8: The instrumentation has been so designed to perform its function over the range of expected environmental conditions associated with its location at the site.

i) Position 9: A plan for the timely utilization of the data obtained from the seismic instrumentation is discussed in Section 3.7.4.4.

3.7.4.2 Location and Description of Instrumentation

In order to comply with regulator,, concerns as to the effect of seismic events on operating plants, a comprehensive program has been prepared for the St. Lucie site. Figure 3.7-40 delineates the location of various instruments provided for measuring the magnitude and spectra of earthquake initiated disturbances. The arrangement is responsive to the guidance provided by the Staff's position. (Question 3.2(b)(c) Staff letter dated 26 April 1974.) However, it is felt that the type of instruments relied upon in the St. Lucie program provide more comprehensive data, and still provide the intelligence required by the operator to initiate protective action. The primary source of comprehensive data is obtained from the time-history accelograph (T/A). ANSI N18.5-1974 states that it is "the basic and most important instrument for measuring vibratory motion."

Of primary importance is that the control room operator be provided with immediate data inputs during a seismic event. Not only must it be known that an event is occurring but the operator must have available sufficient information to determine if the disturbance is of a magnitude which may require shutdown. A communication link with the containment building monitors is available to provide the operator with the peak acceleration over the entire frequency spectrum for three orthogonal axes. Annunciation is provided when the seismic instruments are triggered and at 90 percent of the OBE. The unit will be secured if the OBE is exceeded.

The two T/A's in each building are connected as shown in Figure 3.7-40. Should an earthquake occur, the master switch in the foundation T/A starts the recorder in the higher building elevation upon receipt of a 0.01 g excitation.

This level is annunciated in the control room for operator notification. A communication link with the containment building monitors is available in the control room to provide access to all data recorded by these triaxial accelerometers. Additionally, should the acceleration intensity reach 90 percent of OBE, as detected by the sensor at the foundation, another control room annunciator is activated. The operator has audible indication that he has approached the OBE and as well as, readily available indication of what peak accelerations were actually achieved. It is concluded that sufficient prompt intelligence is available for the operator to make a decision to shutdown.

Turning to the alternate recommendations for control room intelligence via response spectrum recorders (seismoscopes). The information they supply is not as complete as that offered by time history accelographs. Annunciation and indication systems activated by response spectrum recorders are only capable of sensing a few (up to twelve) points in the earthquake frequency domain. Hence, it is possible that the frequencies selected for recording and control room annunciation are not sufficient to determine the true magnitude of earthquake disturbances. This results from,the fact that the frequency band of each sensing element is very narrow. Thus, major seismic components may not be available for operator decision, i.e., peaks occurring between the discrete frequencies monitored could very well go undetected.

In light of the above it is felt that the seismic instruments provided for St. Lucie provide more comprehensive prompt information to the operator than the seismoscope alternative. Turning now to the question of recording data for post-seismic event evaluation, again the T/A's are relied upon. They provide a complete time-history on electronic storage media, whereas the seismoscope provides this data for 12 discrete frequencies. Further discussion of recorded data is provided below.

With regard to peak accelerographs, three will be provided. The location of these instruments will be in accordance with the staff's request (Q3.2(b)(c)), shown on Figure 3.7-40. Implementation will be limited only :by the natural physical capabilities of the instruments., i.e., temperature and/or radiation. Based on Revision 2 of Regulatory Guide 1.12, the original three peak recording accelerographs were deleted.

Although the response spectrum recorders do have sampling limitations, they are capable of supplying fairly timely post earthquake spectrum information, whereas T/A spectral data requires some time for computer analysis. Basically, each of twelve tuning forks scratch their motion on a metal plate. Thus, subsequent to actions initiated by the event, e.g., shutdown, inspections, the operator could obtain the sheet and determine the response at twelve frequencies. One response spectrum recorder will be provided on the containment foundation in the vicinity of the T/A and another inside containment on category 1 piping or equipment support. The operator will then have the ability to compare the peak accelerations with those at the-discrete frequencies sampled by the response spectrum recorders. Based on Revision 2 of Regulatory Guide 1.12, the original response spectrum recorders were deleted.

Besides the seismoscopes provided above, two additional T/A's will be provided in the RAB. One will be located at the foundation and one on an upper level of the building. The upper level T/A will be mounted on category 1 piping or equipment support.

The instruments located in the containment are independent of those located in the reactor auxiliary building.

3.7.4.3 Control Room Operator Notification

All five triaxial strong motion accelerographs (T/As) record accelerations on electronic storage media which can be removed from the machine and interpreted with the aid of computer software. The readout gives accelerations in two horizontal and one vertical direction. In addition to local recording, a communication link with the containment building monitors is available in the Unit 1 control room. The operator can thus obtain the peak magnitude of the seismic disturbance in each of the three axes without leaving the control room. An acceleration of 0.01g triggers both accelerographs causing alarms in both control rooms.

Upon actuation, the accelerographs capture pre-trigger data (length of time is adustable) and will continue to operate for as long as the master detects the earthquake plus a length of time (adjustable) after the event is over. It then resets and is available for the next event.

The value of 90 percent of the OBE used for the alarm setting on the master accelerograph was elected to conservatively account for instrument inaccuracies and to assure that seismic motion approaching critical values is indicated.

3.7.4.4 Comparison of Measured and Predicted Responses

The plant operators are provided with procedures and criteria to review the accelerations recorded at the plant site. These criteria consider system design and dynamic analyses in establishing the acceptable levels for continued operation.

Should a seismic event be realized during plant operation and subsequent data analysis reveals the predicted responses are not within 20 percent of that measured, appropriate action will be taken to correct the dynamic models utilized for St. Lucie.

For earthquakes which have equaled or exceeded the OBE spectrum, the event will be reported to the NRC for evaluation of the required procedures prior to restart of the plant.

Technical Specification Amendment 135 was issued on April 25, 1995 which removed the seismic instrumentation requirements from the Technical Specification and relocated them to the UFSAR. Chapter 13 Section 8.1 provides information that was contained in the Technical Specifications and is provided in accordance with that amendment.

3.7.5 SEISMIC DESIGN CONTROL MEASURES

Purchase specifications for Seismic Class I components specify horizontal and vertical seismic acceleration values based on the floor response spectra at the equipment location. The vendors are required to demonstrate by calculations, operating experience data or testing, the capability of the equipment to withstand the seismic forces, and that the equipment will continue to operate after a DBE. The floor response spectra at the location of the equipment are supplied to the vendors. The vendors were also required to calculate the natural period of vibration of the equipment and see that the natural period does not fall within the critical frequency range of the floor response spectra.

The calculations and/or test results received from the vendors are reviewed by cognizant engineering personnel for approval of the seismic qualification method used and verification that the equipment meets the design conditions specified. The review and approval of vendor seismic qualification data follows the same procedure as approval of drawings and other design data.

REFERENCES FOR SECTION 3.7

1. Hou, Shou-Nien, "Earthquake Simulation Models and Their Applications," Research Report R68-17, Department of Civil Engineering, Massachusetts Institute of Technology, Cambridge, Massachusetts, May 1968.

2. Vanmarcke, Erik H. & C. Allin Cornell, "Analysis of Uncertainty in Ground Motion and Structural Response Due to Earthwuakes," Research Report R69-24, Department of Civil Engineering, Massachusetts Institute of Technology, Cambridge, Massachusetts, April 1969.

3. Whitman, R. V. and F. E. Richart, "Design Procedures for Dynamically Loaded Foundations".

4. Przemieniecki, J. S., "Theory of Matrix Structural Analysis," Chapter 13, McGraw-Hill Book Company, New York, New York, 1968.

5. Hurty, W. C., and Rubinstein, M. F., "Dynamics of Structures," Chapter 8, Prentice-Hall, Inc., Englewood Cliffs, New Jersey, 1964.

6. ICES STRUDL II Engineering Users Manual, R68-91, Department of Civil Engineering, Massachusetts Institute of Technology, Cambridge, Mass.

7. Newmark, N. M., "A Method of Computation for Structural Dynamics," Volume 3, Journal of Engineering Mechanics Division, A.S.C.E., July, 1959.

8. Fritz, R. J., and Kiss, E., "The Vibration Response of a Cantilevered Cylinder Surrounded by an Annular Fluid," KAPL-M-6539, February 1966.

9. Kiss, E., "Analysis of the Fundamental Vibration Frequency of a Radial Vane Internal Steam Generator Structure," ANL-7685, Proceedings of Conference on Flow-Induced Vibrations in Reactor System Components, May 1970, Argonne National Laboratory, Argonne, Ill.

10. Combustion Engineering Test Report, "CEA Full Assembly Mechanical Tests," TR-DT-37, November 17, 1970.

11. "MRI/STARDYNE - Static and Dynamic Structural Analysis System: User Information Manual," Control Data-Corporation, June 1, 1970.

12. MRI/STARDYNE User Manual, Computer Methods Department, Mechanics Research, Inc., Los Angeles, California, January 1, 1970.

13. Gabrielson, V. K., "SHOCK - A Computer Code for Solving Lumped-Mass Dynamic Systems," SCL-DR-65-34, January 1966.

14. Nigam, N.C. and Jennings, P.C., "Digital Calculation of Response Spectra from Strong Motion Earthquake Records," June 1968.

15. Framatome ANP Document 32-5007216-09, "Program Verification for BWSPAN PC Version".

TABLE 3.7-1

REACTOR BUILDING PROPERTIES
HORIZONTAL MODEL

SHIELD BUILDING

Mass No.	W (k)	A (Ft^2)	I (1000 Ft^4)
1	7,081.0	585.2	2,815.0
2	5,251.0	711.2	4,057.0
3	4,107.2	711.2	4,057.0
4	4,267.3	711.2	4,057.0
5	4,267.3	711.2	4,057.0
6	4,267.3	711.2	4,057.0
7	4,267.3	711.2	4,057.0
8	4,167.3	711.2	4,057.0
9	4,267.3	711.2	4,057.0
10	3,200.4	711.2	4,057.0

STEEL CONTAINMENT

Mass No.	W (k)	A (Ft^2)	I (1000 Ft^4)
11	336.3	110.8	293
12	265.5	138.9	571
13	307.5	150.5	726.5
14	508.0	302.5	1486
15	1,196.1	438.9	1807
16	1,196.1	302.5	1486
17	684.5	302.5	1486
18	684.5	302.5	l486
19	684.5	302.5	1486
20	554.4	358.8	1746

INTERNAL STRUCTURE

Mass No.	W (k)	A (Ft^2)	I (1000 Ft^4)
21	615.9	256.0	65.9
22	10,139.0	789.0	2,080.0
23	7,373.0	939.0	2,090.0
24	4,945.0	1,089.0	2,100.0

FOUNDATION

W (k)	Weight Moment Inertia ($Ft\text{-}Sec^2\text{-}k$)
116,000	5.414×10^6

Note: Design concrete strength f_c^1 = 4000 psi

Young's modulus E_c = 481,800 ksf

Poisson's ratio μ = 0.17

TABLE 3.7-1A

SOILS-SUPPORTED
SEISMIC CLASS I STRUCTURES

Structure	Foundation Elevation in Feet	Depth of Engineered Back Fill
Reactor Building	-25	35
Reactor Auxiliary Building	-18	42
Fuel Handling Building	+12	72
Intake Structure	-35	25
Diesel Generator Building	+10	70
Steam Trestle	-5	55
Component Cooling Water Heat Exchange Structure	+10	70
Condensate Storage Tank	+10	70
Refueling Water Tank	+10	70
Diesel Fuel Oil Storage Tanks	+10	70

TABLE 3.7-1B

SIDE SPRING CONSTANTS EFFECT ON SOIL STRUCTURE INTERACTION

	Maximum Acceleration, g		
Magnitude of Side Spring, %	Top Shield Bldg	Top Steel Containment Vessel	Foundation Mat
120	0.3294	0.2613	0.1737
110	0.3294	0.2620	0.1715
100	0.3293	0.2626	0.1692
90	0.3290	0.2631	0.1667
80	0.3285	0.2635	0.1642

TABLE 3.7-2

REACTOR BUILDING PROPERTIES
VERTICAL MODEL

SHIELD BUILDING

Mass No.	W (k)	S (k/ft)
1	9,392.0	$5.7x10^6$
2	9,181.0	$17.1x10^6$
3	8,535.0	$17.1x10^6$
4	8,535.0	$17.1x10^6$
5	7,467.0	$17.1x10^6$

STEEL CONTAINMENT

Mass No.	W (k)	S (k/ft)
6	616.0	$3.24x10^6$
7	1,400.0	$7.29x10^6$
8	2,136.0	$7.29x10^6$
9	1,369.0	$7.29x10^6$
10	886.0	$29.15x10^6$

INTERNAL STRUCTURE

Mass No.	W (k)	S (k/ft)
11	616.0	$29.34x10^6$
12	10,139.0	$78.60x10^6$
13	7,373.0	$24.18x10^7$
14	4,945.0	$20.86x10^7$

FOUNDATION

W (k)
116,000

Note: Design concrete strength f_c^1 = 4000 psi

Young's modulus E_c = 481,800 ksf

Poisson's ratio μ = 0.17

TABLE 3.7-3

REACTOR AUXILIARY BUILDING PROPERTIES
HORIZONTAL MODEL

Mass No.	W (K)	A(N-S) (Ft^2)	I(N-S) (Ft^4)	A(E-W) (Ft^2)	I(E-W) (Ft^4)
1	4,263.5	292.0	1,380,000	436.0	765,400
2	12,978.0	881.4	7,687,000	704.7	3,029,400
3	20,351.0	1,632.4	16,452,164	1,765.1	6,072,617
4	28,666.0	1,689.2	26,358,786	2,263.0	7,548,625
Base	40,517.0		*4,355,415		*1,000,955

* Weight moment of inertia (k-sec ^{2}-ft)

N-S Long direction

E-W Short direction

Design concrete strength = f_c^1 3000 psi

Young's modulus E_c = 422,000 ksf

Poisson's ratio μ = 0.17

Shear modulus G_c = 180,000 ksf

TABLE 3.7-4

REACTOR AUXILIARY BUILDING PROPERTIES
VERTICAL MODEL

Mass No.	W	S (10^3K/ft)	Mass No.	W	S (10^3K/ft)
1	1,127.0	17,440	9	35.7	20.6
2	12,525.4	44,280	10	81.0	24.9
3	18,519.0	78,980	11	191.0	131.0
4	27,568.0	78,200	12	270.0	28.8
5	810.0	15.8	13	207.0	0.758
6	32.2	18.7	14	51.5	183.0
7	74.0	0.304	15	100.0	420.0
8	70.0	16.6	Base	40,517.0	

Design concrete strength f_c^1 = 3000 psi

Young's modulus E_c = 422,000 ksf

Poisson's ratio μ = 0.17

Shear modulus G_c = 180,000 ksf

TABLE 3.7-5

FUEL HANDLING BUILDING PROPERTIES

HORIZONTAL MODEL

Mass No.	A(N-S) (ft^4)	I(N-S) (ft^4)	A(E-W) (ft^2)	I(E-W) (ft^4)	W (k)
1	251.0	859,000	212.0	187,000	3,510.0
2	932.0	1,728,000	877.5	527,000	3,639.0
3	984.0	1,834,000	921.5	531,000	14,103.0
Base		*7,973,100(1)		*1,078,200(1)	7,201(1)
		or 11,078,800(2)		or 1,498,100(2)	or 10,006(2)

* Weight Moment of Inertia ($k\text{-}sec^{2}\text{-}ft$)

N-S Long Direction of Structure

E-W Short Direction of Structure

VERTICAL MODEL

Mass No.	W (k)	S (k/ft)
1	3,510.0	5,950,000
2	3,639.0	54,000,000
3	12,500.0	25,600,000
4	180.0	98,000
5	248.0	33,300
Base	7,201(1)	
	or 10,006(2)	

Design concrete strength f_c^1 = 3000 psi

Young's modulus E_c = 422,000 ksf

Poisson's Ratio μ = 0.17

Shear Modulus G_c = 180,000 ksf

(1) for min fuel rack wt
(2) for max fuel rack wt

TABLE 3.7-6

INTAKE STRUCTURE PROPERTIES
HORIZONTAL MODEL

Mass No.	W (k)	A(N-S) (ft^2)	I(N-S) (ft^4)	A(E-W) (ft^2)	I(E-W) (ft^4)
1	3,451.4	230.7	353,000	704.0	491,400
2	3,923.7	259.5	364,500	605.0	322,340
3	4,250.0	202.0	320,000	728.0	340,000
Base	4,242.0		*38,854		*36,600

* Weight Moment of Inertia ($k\text{-}sec^2\text{-}ft$)

N-S Long Direction

E-W Short Direction

Design Concrete Strength f_c^1 = 4,000 psi

Young's Modulus E_c = 481,000 ksf

Poisson's Ratio μ = 0.17

Shear Modulus G_c = 206,000 ksf

VERTICAL MODEL

Mass No.	W (k)	S (k/ft)
1	3,350.9	2.90×10^7
2	3,923.7	2.92×10^7
3	4,250.0	3.15×10^7
4	41.2	1.52×10^6

TABLE 3.7-7

DIESEL GENERATOR BUILDING PROPERTIES
HORIZONTAL MODEL

Mass No.	W (k)	A(N-S) (ft^2)	I(N-S) (ft^4)	A(E-W) (ft^2)	I(E-W) (ft^4)
1	1,598.0	418.0	557,000	224.0	465,000
2	1,282.0	418.0	557,000	224.0	465,000
3	2,047.0	418.0	557,000	224.0	465,000
4	1,512.0	532.0	255,000	350.0	2,175
5	1,512.0	532.0	255,000	350.0	2,175
Base	3,493.0		*144,000		*82,300

* Weight Moment of Inertia (k-sec^2-ft)

N-S Long Direction

E-W Short Direction

Design Concrete Strength f_c^1 = 3,000 psi

Young's Modulus E_c = 422,000 ksf

Poisson's Ratio μ = 0.17

Shear Modulus G_c = 180,000 psf

VERTICAL MODEL

Mass No.	W (k)	S (k/ft)
1	640.0	24,600,000
2	1,282.0	24,600,000
3	1,135.0	32,000,000
4	380.0	213,200
5	600.0	22,500,000
6	600.0	22,500,000
Base	7,141.0	

TABLE 3.7-8

COMPARISON OF STRUCTURAL RESPONSES
FOR RESPONSE SPECTRA AND TIME HISTORY SEISMIC
ANALYSIS METHODS

N-S Direction (0.1 G Horizontal Earthquake)

	Response	
React. Aux. Bldg. Elevation	Response Spectra Method	Time History Method
82.0	0.382g	0.40g
62.0	0.345g	0.28g
43.0	0.292g	0.23g
19.5	0.214g	0.23g
- 0.5	0.134g	0.30g

E-W DIRECTION (0.1 G HORIZONTAL EARTHQUAKE)

	Response	
React. Aux. Bldg. Elevation	Response Spectra Method	Time History Method
82.0	0.360g	0.52g
62.0	0.325g	0.40g
43.0	0.279g	0.36g
19.5	0.219g	0.34g
- 0.5	0.166g	0.34g

TABLE 3.7-9

REACTOR BUILDING NATURAL PERIODS OF VIBRATION (SEC)

HORIZONTAL

MODE	E=48000 psi	E=44000 psi	E=40000 psi	E=36000 psi	E=32000 psi
1	0.6766	0.7046	0.7373	0.7740	0.8180
2	0.2725	0.2840	0.2971	0.3221	0.3302
3	0.1365	0.1370	0.1375	0.1379	0.1383
4	0.0879	0.0881	0.0884	0.0886	0.0888
5	0.0716	0.0716	0.0716	0.0716	0.0716
6	0.0492	0.0493	0.0493	0.0494	0.0494
7	0.0421	0.0421	0.0421	0.0421	0.0421
8	0.0383	0.0383	0.0383	0.0383	0.0383
9	0.0283	0.0283	0.0283	0.0283	0.0283
10	0.0232	0.0232	0.0232	0.0232	0.0232

VERTICAL

MODE	E=48000 psi	E=44000 psi	E=40000 psi	E=36000 psi	E=32000 psi
1	0.4447	0.4647	0.4876	0.5144	0.5461
2	0.0828	0.0828	0.0828	0.0829	0.0829
3	0.0434	0.0434	0.0434	0.0435	0.0435
4	0.0379	0.0379	0.0379	0.0379	0.0379
5	0.0227	0.0227	0.0227	0.0227	0.0227
6	0.0170	0.0170	0.0170	0.0170	0.0170
7	0.0164	0.0164	0.0164	0.0164	0.0164
8	0.0155	0.0155	0.0155	0.0155	0.0155
9	0.0129	0.0129	0.0129	0.0129	0.0129
10	0.0112	0.0112	0.0112	0.0112	0.0112

TABLE 3.7-10

REACTOR BUILDING HORIZONTAL

STRUCTURAL RESPONSES

Mass No.	E = 48000 psi A (g)	E = 48000 psi D (ft)	E = 44000 psi A (g)	E = 44000 psi D (ft)	E = 40000 psi A (g)	E = 40000 psi D (ft)	E = 36000 psi A (g)	E = 36000 psi D(ft)	E = 32000 psi A (g)	E = 32000 psi D (ft)	Envelop A (g)	Envelop D (ft)
1	.3459	.1052	.3390	.1112	.3322	.1189	.3241	.1276	.3115	.1360	.3459	.1360
2	.3006	.0951	.2945	.1006	.2883	.1075	.2813	.1155	.2700	.1230	.3006	.1230
3	.2653	.0871	.2598	.0922	.2542	.0985	.2480	.1058	.2378	.1128	.2653	.1128
4	.2228	.0771	.2181	.0816	.2133	.0873	.2081	.0938	.1993	.1000	.2228	.1000
5	.1888	.0683	.1847	.0724	.1807	.0774	.1763	.0833	.1687	.0888	.1888	.0888
6	.1599	.0596	.1564	.0632	.1530	.0676	.1494	.0728	.1428	.0777	.1599	.0777
7	.1390	.0509	.1360	.0540	.1331	.0579	.1301	.0624	.1245	.0667	.1390	.0667
8	.1292	.0424	.1267	.0451	.1243	.0484	.1215	.0522	.1168	.0559	.1292	.0559
9	.1314	.0341	.1294	.0364	.1275	.0392	.1249	.0423	.1208	.0455	.1314	.0455
10	.1433	.0263	.1418	.0282	.1403	.0304	.1378	.0330	.1342	.0356	.1433	.0356
11	.2665	.0914	.2639	.0975	.2619	.1052	.2597	.1140	.2533	.1226	.2665	.1226
12	.2402	.0839	.2372	.0896	.2347	.0466	.2321	.1047	.2258	.1126	.2402	.1126
13	.2169	.0770	.2135	.0822	.2107	.0887	.2078	.0961	.2015	.1033	.2169	.1033
14	.1935	.0695	.1897	.0742	.1864	.0801	.1832	.0868	.1769	.0933	.1935	.0933
15	.1731	.0621	.1690	.0664	.1653	.0716	.1617	.0775	.1553	.0834	.1731	.0884
16	.1564	.0548	.1521	.0585	.1481	.0631	.1442	.0683	.1378	.0735	.1564	.0735
17	.1440	.0474	.1398	.0506	.1359	.0546	.1319	.0591	.1258	.0636	.1440	.0636
18	.1378	.0400	.1341	.0428	.1306	.0462	.1268	.0500	.1212	.0539	.1378	.0539
19	.1385	.0329	.1356	.0351	.1329	.0379	.1295	.0411	.1247	.0445	.1385	.0443
20	.1457	.0259	.1439	.0278	.1422	.0301	.1394	.0326	.1356	.0352	.1457	.0352
21	.1389	.0398	.1353	.0426	.1320	.0460	.1281	.0499	.1225	.0537	.1389	.0537
22	.1392	.0369	.1359	.0395	.1327	.0426	.1288	.0462	.1234	.0498	.1392	.0498
23	.1416	.0313	.1389	.0335	.1363	.0362	.1328	.0393	.1281	.0424	.1416	.0424
24	.1486	.0246	.1470	.0264	.1455	.0285	.1427	.0310	.1390	.0355	.1486	.0335
Base	.1680	.0170	.1675	.0184	.1672	.0199	.1655	.0217	.1625	.0236	.1680	.0236

TABLE 3.7-11

REACTOR BUILDING VERTICAL
STRUCTURAL RESPONSES

Mass No.	E = 48000 psi A (g)	E = 48000 psi D (ft)	E = 44000 psi A (g)	E = 44000 psi D (ft)	E = 40000 psi A (g)	E = 40000 psi D (ft)	E = 36000 psi A (g)	E = 36000 psi D (ft)	E = 32000 psi A (g)	E = 32000 psi D (ft)	Envelop A (g)	Envelop D (ft)
1	.1363	.0220	.1331	.0234	.1296	.0251	.1254	.0270	.1203	.0292	.1363	.0292
2	.1349	.0218	.1318	.0232	.1285	.0249	.1244	.0268	.1194	.0290	.1349	.0290
3	.1340	.0216	.1309	.0231	.1278	.0248	.1237	.0267	.1189	.0289	.1340	.0289
4	.1326	.0214	.1297	.0229	.1267	.0246	.1228	.0265	.1181	.0287	.1326	.0287
5	.1309	.0211	.1282	.0226	.1253	.0243	.1216	.0262	.1171	.0285	.1309	.0285
6	.1306	.0210	.1278	.0225	.1248	.0242	.1213	.0262	.1168	.0284	.1304	.0284
7	.1303	.0210	.1276	.0225	.1246	.0242	.1212	.0262	.1167	.0284	.1303	.0284
8	.1300	.0210	.1274	.0224	.1243	.0242	.1210	.0261	.1166	.0284	.1300	.0284
9	.1296	.0209	.1270	.0224	.1238	.0241	.1208	.0261	.1163	.0283	.1296	.0283
10	.1290	.0208	.1265	.0223	.1239	.0240	.1203	.0260	.1160	.0282	.1290	.0282
11	.1291	.0208	.1266	.0223	.1239	.0240	.1204	.0260	.1160	.0282	.1291	.0282
12	.1290	.0208	.1265	.0223	.1238	.0240	.1204	.0260	.1160	.0282	.1290	.0282
13	.1290	.0208	.1264	.0223	.1238	.0240	.1203	.0260	.1160	.0282	.1290	.0282
14	.1289	.0208	.1264	.0223	.1237	.0240	.1202	.0259	.1159	.0282	.1289	.0282
Base	.1288	.0208	.1263	.0223	.1237	.0240	.1202	.0259	.1158	.0282	.1288	.0282

TABLE 3.7-12

REACTOR AUXILIARY BUILDING
NATURAL PERIODS OF VIBRATION (SEC)
HORIZONTAL

	E=48000 psi		E=44000 psi		E=40000 psi		E=36000 psi		E=32000 psi	
Mode	N-S	E-W	N-S	E-W	N-S	E-W	N-S	E-W	N-S	E-W
1	.4453	.7326	.4645	.7649	.4868	.8020	.5125	.8452	.5421	.8952
2	.2167	.1923	.2263	.2009	.2375	.2107	.2503	.2220	.2653	.2355
3	.0572	.0469	.0573	.0469	.0574	.0469	.0574	.0469	.0575	.0470
4	.0437	.0348	.0438	.0348	.0438	.0348	.0438	.0348	.0438	.0348
5	.0335	.0289	.0335	.0289	.0335	.0289	.0335	.0289	.0335	.0289
6	.0266	.0262	.0266	.0262	.0266	.0262	.0266	.0262	.0266	.0262

VERTICAL

Mode	E=48000 psi	E=44000 psi	E=40000 psi	E=36000 psi	E=32000 psi
1	0.5788	0.5788	0.5789	0.5789	0.5790
2	0.5464	0.5464	0.5464	0.5464	0.5464
3	0.2901	0.3017	0.3158	0.3324	0.3515
4	0.2469	0.2476	0.2481	0.2485	0.2489
5	0.1071	0.1071	0.1071	0.1071	0.1071
6	0.0719	0.0719	0.0719	0.0719	0.0719
7	0.0631	0.0631	0.0631	0.0631	0.0631
8	0.0461	0.0461	0.0461	0.0461	0.0461

TABLE 3.7-13

REACTOR AUXILIARY BUILDING HORIZONTAL
STRUCTURAL RESPONSES

Mass No.	E = 48000 psi A (g)	E = 48000 psi D (ft)	E = 44000 psi A (g)	E = 44000 psi D(ft)	E = 40000 psi A (g)	E = 40000 psi D (ft)	E = 36000 psi A (g)	E = 36000 psi D (ft)	E = 32000 psi A (g)	E = 32000 psi D (ft)	Envelop A (g)	Envelop D (ft)
1	.2669	.0418	.2614	.0445	.2557	.0478	.2485	.0514	.2402	.0554	.2669	.0554
2	.2170	.0350	.2124	.0372	.2076	.0400	.2015	.0430	.1944	.0464	.2170	.0464
3	.1788	.0287	.1751	.0305	.1713	.0328	.1663	.0353	.1606	.0381	.1788	.0381
4	.1494	.0207	.1472	.0220	.1449	.0237	.1420	.0255	.1386	.0276	.1494	.0276
Base	.1543	.0120	.1541	.0129	.1538	.0139	.1535	.0152	.1528	.0166	.1543	.0166

E-W

Mass No.	E = 48000 psi A (g)	E = 48000 psi D (ft)	E = 44000 psi A (g)	E = 44000 psi D (ft)	E = 40000 psi A (g)	E = 40000 psi D (ft)	E = 36000 psi A (g)	E = 36000 psi D (ft)	E = 32000 psi A (g)	E = 32000 psi D (ft)	Envelop A(g)	Envelop D(ft)
1	.2455	.0956	.2415	.1019	.2363	.1091	.2239	.1131	.2114	.1176	.2455	.1176
2	.1792	.0762	.1755	.0812	.1711	.0869	.1603	.0901	.1494	.0937	.1792	.0937
3	.1346	.0581	.1317	.0619	.1284	.0663	.1201	.0687	.1116	.0715	.1346	.0715
4	.1297	.0348	.1295	.0371	.1281	.0398	.1250	.0413	.1219	.0430	.1297	.0430
Base	.1900	.0095	.1926	.0127	.1927	.0111	.1928	.0119	.1929	.0128	.1929	.0128

TABLE 3.7-14

REACTOR AUXILIARY BUILDING
VERTICAL STRUCTURAL RESPONSES

Mass	E = 48000 psi		E = 44000 psi		E = 40000 psi		E = 36000 psi		E = 32000 psi		Envelop	
No.	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)
1	.1366	.0094	.1404	.0104	.1412	.0115	.1408	.0127	.1396	.0141	.1412	.0141
2	.1361	.0093	.1400	.0104	.1409	.0115	.1405	.0127	.1394	.0141	.1409.	.0141
3	.1354	.0093	.1393	.0103	.0402	.0114	.1400	.0126	.1389	.0140	.1402	.0140
4	.1345	.0092	.1384	.0103	.1395	.0113	.1393	.0126	.1383	.0139	.1395	.0139
5	.6643	.0417	.5472	.0370	.4508	.0333	.3769	.0309	.3209	.0296	.6643	.0417
6	.1398	.0096	.1435	.0106	.1442	.0117	.1435	.0129	.1421	.0143	.1442	.0143
7	.1676	.0389	.1764	.0405	.1873	.0426	.2017	.0454	.2213	.0493	.2213	.0493
8	.1457	.0100	.1490	.0110	.1492	.0121	.1480	.0133	.1461	.0147	.1492	.0147
9	.1391	.0095	.1428	.0106	.1435	.0117	.1429	.0129	.1416	.0143	.1435	.0143
10	.1425	.0098	.1459	.0108	.1463	.0119	.1455	.0131	.1438.	.0145	.1463	.0145
11	.1383	.0095	.1420	.0105	.1428	.0116	.1423	.0128	.1409	.0142	.1428	.0142
12	.1583	.0108	.1605	.0118	.1594	.0129	.1569	.0141	.1537	.0154	.1605	.0154
13	.1557	.0409	.1619	.0421	.1698	.0438	.1800	.0459	.1945	.0490	.1945	.0490
14	.1351	.0093	.1390	.0103	.1400	.0114	.1397	.0126	.1387	.0140	.1400	.0140
15	.1349	.0092	.1388	.0103	.1398	.0114	.1396	.0126	.1387.	0140	.1398	.0140
Base	.1329	.0091	.1369	.0102	.1381	.0112	.1381	.0124	.1372	.0138	.1381	.0138

TABLE 3.7-15

FUEL HANDLING BUILDING
NATURAL PERIODS OF VIBRATION (SEC)
HORIZONTAL

Mode	E=42000 psi		E=38500 psi		E=35000 psi		E=31500 psi		E=2800 psi	
	N-S	E-W	N-S	E-W	N-S	E-W	N-S	E-W	N-S	E-W
1	.4649	.5303	.4854	.5533	.5090	.5804	.5363	.6111	.5687	.6475
2	.2450	.1931	.2351	.2015	.2673	.2112	.2812	.2222	.2977	.2355
3	.0592	.0665	.0593	.0666	.0593	.0666	.0594	.0666	.0594	.0667
4	.0303	.0313	.0304	.0314	.0304	.0314	.0305	.0315	.0305	.0315
5	.0166	.0178	.0166	.0178	.0166	.0178	.0166	.0178	.0166	.0178

VERTICAL

Mode	E=42000 psi	E=38500 psi	E=35000 psi	E=31500 psi	E=2800 psi
1	0.2440	0.2547	0.2672	0.2813	0.2982
2	0.0950	0.0951	0.0951	0.0951	0.0951
3	0.0473	0.0473	0.0473	0.0473	0.0473
4	0.0267	0.0267	0.0267	0.0267	0.0267
5	0.0141	0.0141	0.0141	0.0141	0.0141
6	0.0075	0.0075	0.0075	0.0075	0.0075

TABLE 3.7-16

FUEL HANDLING BUILDING HORIZONTAL
STRUCTURAL RESPONSES

N-S

Mass	E = 42000 psi		E = 38500 psi		E = 35000 psi		E = 31500 psi		E = 28000 psi		Envelop	
No.	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)
1	.1637	.0158	.1621	.0168	.1602	.0180	.1582	.0194	.1561	.0211	.1637	.0211
2	.1719	.0122	.1716	.0130	.1711	.0141	.1706	.0153	.1700	.0169	.1719	.0169
3	.1790	.0111	.1790	.0119	.1789	.0130	.1787	.0142	.1786	.0157	.1790	.0157
Base	.1942	.0097	.1952	.0106	.1962	.0117	.1972	.0130	.1983	.0146	.1983	.0146

E-W

Mass	E = 42000 psi		E = 39500 psi		E = 35000 psi		E = 31500 psi		E = 28000 psi		Envelop	
No.	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)
1	.2003	.0457	.1947	.0484	.1886	.0515	.1822	.0552	.1756	.0597	.2003	.0597
2	.1423	.0286	.1397	.0303	.1364	.0323	.1331	.0347	.1295	.0375	.1423	.0375
3	.1404	.0218	.1396	.0231	.1378	.0247	.1360	.0265	.1340	.0287	.1404	.0287
Base	.1832	.0075	.1860	.0081	.1867	.0083	.1873	.0097	.1880	.0107	.1880	.0107

TABLE 3.7-17

FUEL HANDLING BUILDING
VERTICAL STRUCTURAL RESPONSES

Mass No.	E = 42000 psi A (g)	E = 42000 psi D (ft)	E = 38500 psi A (g)	E = 38500 psi D (ft)	E = 35000 psi A (g)	E = 35000 psi D (ft)	E = 31500 psi A (g)	E = 31500 psi D (ft)	E = 28000 psi A (g)	E = 28000 psi D (ft)	Envelop A (g)	Envelop D (ft)
1	.1486	.0072	.1485	.0079	.1483	.0086	.1481	.0096	.1479	.0107	.1486	.0107
2	.1468	.0071	.1468	.0078	.1467	.0085	.1467	.0095	.1467	.0106	.1468	.0106
3	.1464	.0071	.1464	.0077	.1464	.0085	.1464	.0094	.1464	.0106	.1464	.0106
4	.1522	.0074	.1517	.0080	.1512	.0088	.1507	.0097	.1503	.0109	.1522	.0109
5	.1738	.0084	.1711	.0090	.1685	.0098	.1660	.0107	.1636	.0118	.1738	.0118
Base	.1443	.0070	.1445	.0076	.1447	.0084	.1448	.0093	.1450	.0105	.1450	.0105

TABLE 3.7-18

INTAKE STRUCTURE NATURAL PERIODS OF VIBRATION (SEC)

HORIZONTAL

Mode	E=48000 psi		E=44000 psi		E=40000 psi		E=36000 psi		E=32000 psi	
	N-S	E-W	N-S	E-W	N-S	E-W	N-S	E-W	N-S	E-W
1	.2782	.2824	.2903	.2946	.3043	.3086		.3248	.3392	
2	.1025	.0999	.1069	.1042	.1120	.1092		.1150	.1245	
3	.0294	.0190	.0295	.0191	.0296	.0191		.0192	.0297	
4	.0253	.0163	.0255	.0164	.0257	.0164		.0165	.0262	
5	.0206	.0124	.0206	.0125	.0206	.0125		.0125	.0207	

VERTICAL

Mode	E=48000 psi	E=44000 psi	E=40000 psi	E=36000 psi	E=32000 psi
1	0.2550	0.2662	0.2791	0.2941	0.3119
2	0.0165	0.0165	0.0165	0.0165	0.0165
3	0.0090	0.0090	0.0090	0.0090	0.0090
4	0.0070	0.0070	0.0070	0.0070	0.0070
5	0.0057	0.0057	0.0057	0.0057	0.0057

TABLE 3.7-19

INTAKE STRUCTURE STRUCTURAL RESPONSES

HORIZONTAL N-S

Mass No.	E = 48000 psi		E = 44000 psi		E = 40000 psi		E = 36000 psi		E = 32000 psi		Envelop	
	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)
1	.2987	.0186	.2987	.0202	.2976	.0221	-	-	.2904	.0267	.2987	.0267
2	.2085	.0131	.2087	.0143	.2081	.0157	-	-	.2030	.0190	.2087	.0190
3	.1509	.0087	.1519	.0096	.1525	.0106	-	-	.1522	.0129	.1525	.0129
Base	.1231	.0038	.1262	.0042	.1295	.0046	-	-	.1360	.0058	.1360	.0058

HORIZONTAL E-W

Mass No.	E = 48000 psi		E = 44000 psi		E = 40000 psi		E = 36000 psi		E = 32000 psi		Envelop	
	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)
1	.2934	.0188	.2934	.0205	.2914	.0223	.2879	.0244	-	-	.2934	.0244
2	.2223	.0145	.2223	.0157	.2209	.0172	.2183	.0188	-	-	.2223	.0188
3	.1720	.0109	.1725	.0119	.1721	.0130	.1707	.0143	-	-	.1725	.0143
Base	.1303	.0068	.1320	.0075	.1334	.0082	.1347	.0091	-	-	.1347	.0091

VERTICAL

Mass No.	E = 48000 psi		E = 44000 psi		E = 40000 psi		E = 36000 psi		E = 32000 psi		Envelop	
	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)	A (g)	D (ft)
1	.1477	.0078	.1476	.0085	.1475	.0094	.1474	.0104	.1460	.0116	.1477	.0116
2	.1473	.0078	.1473	.0085	.1472	.0094	.1471	.0104	.1457	.0116	.1473	.0116
3	.1466	.0077	.1466	.0085	.1466	.0093	.1466	.0103	.1453	.0115	.1466	.0115
4	.1477	.0078	.1476	.0085	.1476	.0094	.1475	.0104	.1460	.0116	.1477	.0116
Base	.1456	.0077	.1457	.0084	.1458	.0093	.1459	.0103	.1446	.0115	.1459	.0115

TABLE 3.7-20

DIESEL GENERATOR BUILDING
NATURAL PERIODS OF VIBRATION (SEC)
HORIZONTAL

Mode	E=36000 psi N-S	E=36000 psi E-W	E=33000 psi N-S	E=33000 psi E-W	E=30000 psi N-S	E=30000 psi E-W	E=27000 psi N-S	E=27000 psi E-W	E=24000 psi N-S	E=24000 psi E-W
1	.1915	.1969	.1997	.2053	.2095	.2150	.2205	.2262	.2336	.2396
2	.0928	.0915	.0969	.0952	.1016	.0997	.1070	.1049	.1134	.1111
3	.0282	.0377	.0282	.0378	.0282	.0379	.0283	.0380	.0283	.0381
4	.0150	.0311	.0150	.0311	.0150	.0311	.0150	.0311	.0150	.0311
5	.0141	.0223	.0141	.0223	.0141	.0223	.0141	.0223	.0141	.0224
6	.0101	.0171	.0101	.0171	.0101	.0171	.0101	.0171	.0101	.0172

VERTICAL

Mode	E=36000 psi	E=33000 psi	E=30000 psi	E=27000 psi	E=24000 psi
1	0.1363	-	0.1492	0.1572	0.1666
2	0.0458	-	0.0459	0.0459	0.0459
3	0.0115	-	0.0115	0.0115	0.0115
4	0.0057	-	0.0057	0.0057	0.0057
5	0.0053	-	0.0053	0.0053	0.0053
6	0.0046	-	0.0046	0.0046	0.0046

TABLE 3.7-21

DIESEL GENERATOR BUILDING
STRUCTURAL RESPONSES

	Mass No.	E = 36000 psi A (g)	D (ft)	E = 33000 psi A (g)	D (ft)	E = 30000 psi A (g)	D (ft)	E = 27000 psi A (g)	D (ft)	E = 24000 psi A (g)	D (ft)	Envelop A (g)	D (ft)
N-S	1	.2646	.0079	.2678	.0087	.2673	.0095	.2668	.0105	.2664	.0118	.2678	.0118
	2	.2336	.0070	.2365	.0077	.2362	.0084	.2359	.0093	.2355	.0105	.2365	.0105
	3	.2024	.0060	.2052	.0067	.2054	.0073	.2054	.0081	.2054	.0091	.2054	.0091
	4	.1983	.0059	.2015	.0065	.2020	.0072	.2023	.0080	.2026	.0090	.2026	.0090
	5	.1983	.0059	.2015	.0065	.2020	.0072	.2023	.0080	.2026	.0090	.2026	.0090
	Base	.1756	.0052	.1784	.0057	.1791	.0063	.1796	.0070	.1801	.0079	.1801	.0079
E-W	1	.2873	.0090	.2880	.0098	.2873	.0107	.2865	.0118	.2859	.0132	.2880	.0132
	2	.2429	.0077	.2437	.0084	.2432	.0092	.2427	.0101	.2421	.0113	.2437	.0113
	3	.1983	.0063	.1994	.0068	.1995	.0075	.1996	.0083	.1997	.0093	.1997	.0093
	4	.1941	.0061	.1956	.0067	.1960	.0074	.1965	.0082	.1969	.0092	.1969	.0092
	5	.1941	.0061	.1956	.0067	.1960	.0074	.1965	.0082	.1969	.0092	.1969	.0092
	Base	.1608	.0050	.1626	.0054	.1634	.0060	.1645	.0067	.1655	.0095	.1655	.0075
Vertical	1	.1251	.0019	-	-	.1334	.0024	.1359	.0027	.1385	.0031	.1385	.0031
	2	.1247	.0019	-	-	.1330	.0024	.1356	.0027	.1382	.0031	.1382	.0031
	3	.1239	.0019	-	-	.1323	.0024	.1350	.0027	.1377	.0034	.1377	.0034
	4	.1421	.0021	-	-	.1481	.0027	.1492	.0030	.1505	.0031	.1502	.0031
	5	.1233	.0019	-	-	.1317	.0024	.1344	.0027	.1372	.0031	.1372	.0031
	6	.1233	.0019	-	-	.1317	.0024	.1344	.0027	.1372	.0031	.1372	.0031
	Base	.1230	.0019	-	-	.1315	.0024	.1342	.0027	.1370	.0031	.1370	.0031

TABLE 3.7-21A

COMPARISON BETWEEN FRQA AND STARDYNE*

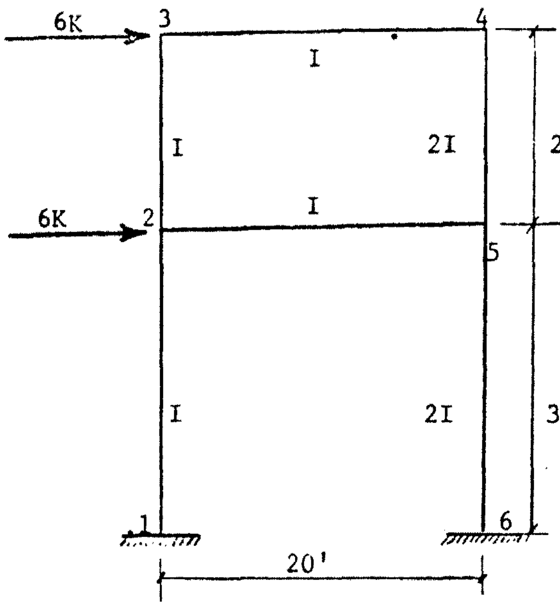
MODE OR MASS PT.	FREQUENCY		DISPLACEMENT AT $T = 1.98_{s=0}$	
	EBASCO DYNAMIC ANALYSIS PROG.	STARDYNE	EBASCO DYNAMIC ANALYSIS PROG.	STARDYNE
1	2.4373	2.4346	0.01631	0.01635
2	5.0158	5.0149	0.01568	0.01572
3	8.3874	8.3599	0.01154	0.01458
4	8.9140	8.9068	0.01368	0.01372
5	11.3003	11.2466	0.01319	0.01323
6	13.1218	13.1031	0.01227	0.01231
7	15.5296	15.4381	0.01091	0.01093
8	17.7919	17.7128	0.00979	0.00982
9	18.4370	18.3135	0.00885	0.00887
10	22.1156	22.0018	0.00781	0.00782
11	23.9480	23.8722	0.01484	0.01488
12	26.0547	25.9860	0.01448	0.01452
13	28.0716	27.9681	0.01412	0.01417
14	30.7635	30.5644	0.01398	0.01402
15	33.2434	33.1839	0.01366	0.01370
16	36.2133	36.0596	0.01319	0.01323
17	37.0848	36.8921	0.01262	0.01265
18	38.5026	38.2458	0.01207	0.01210
19	43.8437	43.6126	0.01153	0.01156

* Refer to Figure 3.7-11A for the representative model

TABLE 3.7-21B

COMPARISON BETWEEN RFRM AND OTHER REFERENCES

A. Sample Problem No. 1: Reference - Structural Theory
Sutherland & Bowman
Page 264



moment sign convention:

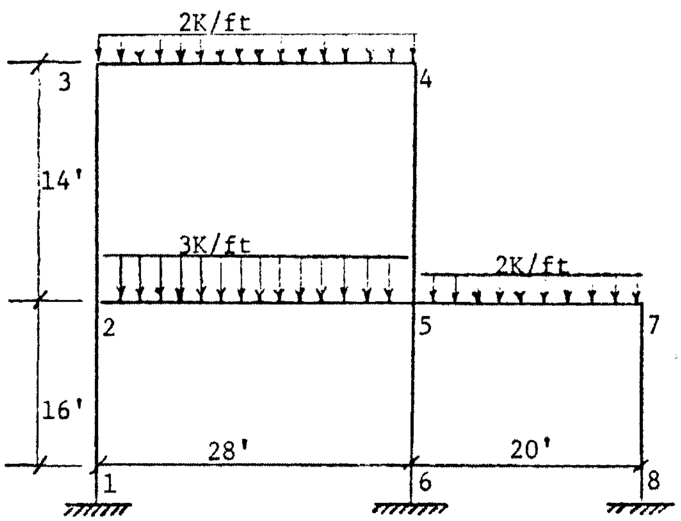
		RFRM RESULTS			**REFERENCE RESULTS**
FOR THE MEMBER 3 4					
	FX(3) =	2.61907	KIPS		
	FY(3) =	-4.22005	KIPS		
	M (3) =	40.04487	FT-K		M(3) = 39.7 FT-K
	FX(4) =	-2.61907	KIPS		
	FY(4) =	4.22005	KIPS		
	M (4) =	44.35622	FT-K		M(4) = 43.8 FT-K
	AXIAL FORCE =	-2.619	KIPS		
FOR THE MEMBER 2 5					
	FX(2) =	4.90394	KIPS		
	FY(2) =	-9.02556	KIPS		
	M (2) =	85.26272	FT-K		M(2) = 84.5 FT-K
	FX(5) =	-4.90394	KIPS		
	FY(5) =	9.02656	KIPS		
	M (5) =	95.26841	FT-K		M(5) = 94.7 FT-K
	AXIAL FORCE =	-4.904	KIPS		

TABLE 3.7-21B (Cont'd)

RFRM RESULTS					REFERENCE RESULT
FOR THE MEMBER 3 2					
	FX(3) =	3.38093	KIPS		
	FY(3) =	4.22005	KIPS		
	M (3) =	-40.04487	FT-K		M(3) = 39.7 FT-K
	FX(2) =	-3.38093	KIPS		
	FY(2) =	-4.22005	KIPS		
	M (2) =	-27.57370	FT-K		M(2) = -26.7 FT- K
	AXIAL FORCE =	4.220	KIPS		
FOR THE MEMBER 4 5					
	FX(4) =	2.61987	KIPS		
	FY(4) =	-4.88005	KIPS		
	M (4) =	-44.38622	FT-K		M(4) = -43.8 FT- K
	FX(5) =	-2.61907	KIPS		
	FY(5) =	-.22005	KIPS		
	M (5) =	-5.02521	FT-K		M(5) = -5.7 FT-K
	AXIAL FORCE =	-4.220	KIPS		
FOR THE MEMBER 2 1					
	FX(2) =	4.47689	KIPS		
	FY(2) =	13.24661	KIPS		
	M (2) =	-57.58-02	FT-K		M(2) - -57.8 FT- K
	FX(1) =	-4.47683	KIPS		
	FY(1) =	-13.24661	KIPS		
	M (1) =	-76.62087	FT-K		M(1) = -75.0 FT- K
	AXIAL FORCE =	-13.247	KIPS		
FOR THE MEMBER 5 6					
	FX(5) =	7.52301	KIPS		
	FY(5) =	-13.24661	KIPS		
	M (5) =	-87.24319	FT-K		M(5) = -87.0 FT- K
	FX(6) -	-7.52301	KIPS		
	FY(6) =	13.24661	KIPS		
	M (6) =	-138.44721	FT-K		M(6) =-136.4 FT- K
	AXIAL FORCE =	-13.247	KIPS		

TABLE 3.7-21B (Cont'd)

B. Sample Problem No. 2: (CASE A) Reference - Reinforced Concrete Structure Peabody Pages 388 to 393



moment sign convention: ↻+

RFRM RESULTS				REFERENCE RESULTS
FOR THE MEMBER 3 4				
	FX(3) =	12.09692	KIPS	
	FY(3) =	27.87728	KIPS	
	M (3) =	-96.73661	FT-K	M(3) = -97.0 FT-K
	FX(4) =	-12.09692	KIPS	
	FY(4) =	28.12272	KIPS	
	M (4) =	100.17281	FT-K	M(4) = 100.8 FT-K
	AXIAL FORCE =	-12.097	KIPS	
FOR THE MEMBER 2 5				
	FX(2) =	-7.05437	KIPS	
	FY(2) =	39.74361	KIPS	
	M (2) =	-125.91879	FT-K	M(2) =-122.1 FT-K
	FX(5) =	7.05437	KIPS	
	FY(5) =	44.25639	KIPS	
	M (5) =	189.09767	FT-K	M(5) = 197.1 FT-K
	AXIAL FORCE =	7.054	KIPS	

TABLE 3.7-21B (Cont'd)

RFRM RESULTS				REFFRENCE RESULT
FOR THE MEMBER' 5 7				
	FX(5) =	2.75273	KIPS	
	FY(5) =	23.28177	KIPS	
	M (5) =	-95.30735	FT-K	M(5) =-103.9 FT-K
	FX(7) =	-2.75273	KIPS	
	FY(7) =	16.71823	KIPS	
	M (7) =	29.67202	FT-K	M(7) = 24.5 FT-K
	AXIAL FORCE =	-2.753	KIPS	
FOR THE MEMBER 3 2				
	FX(3) =	-12.09692	KIPS	
	FY(3) =	-27.87728	KIPS	
	M (3) =	96.73661	FT-K	M(3) = 97.0 FT-K
	FX(2) =	12.09692	KIPS	
	FY(2) =	27.87728	KIPS	
	M (2) =	72.62025	FT-K	M(2) = 73.5 FT-K
	AXIAL FORCE =	-27.877	KIPS	
FOR THE MEMBER 2 1				
	FX(2) =	-5.04255	KIPS	
	FY(2) =	-67.62089	KIPS	
	M (2) =	53.29854	FT-K	M(2) = 48.7 FT-K
	FX(1) =	5.04255	KIPS	
	FY(1) =	67.62089	KIPS	
	M (1) =	27.38224	FT-K	M(1) = 24.3 FT-K
	AXIAL FORCE =	-67.621	KIPS	
FOR THE MEMBER 4 5				
	FX(4) =	12.09692	KIPS	
	FY(4) =	-28.12272	KIPS	
	M (4) =	-100.17281	FT-K	M(4) =-100.8 FT-K
	FX(5) =	-12.09692	KIPS	
	FY(5) =	28.12272	KIPS	
	M (5) =	-69.18404	FT-K	M(5) = -69.1 FT-K
	AXIAL FORCE =	28.123	KIPS	

TABLE 3.7-21B (Cont'd)

RFRM RESULTS					REFERENCE RESULT
FOR THE MEMBER 5 6					
	FX(5) =	2.28982	KIPS		
	FY(5) =	-95.66088	KIPS		
	M (5) =	-24.60627	FT-K		M(5) = -24.1 FT-K
	FX(6) =	-2.28982	KIPS		
	FY(6) =	95.66088	KIPS		
	M (5) =	-12.03081	FT-K		M(6) = -12.1 FT-K
	AXIAL FORCE =	-95.661	KIPS		
FOR THE MEMBER 7 8					
	FX(7) =	2.75273	KIPS		
	FY(7) =	-16.71823	KIPS		
	M (7) =	-29.67202	FT-K		M(7) = -24.5 FT-K
	FX(8) =	-2.75273	KIPS		
	FY(8) =	16.71823	KIPS		
	M (8) =	-14.37167	FT-K		M(8) = -12.3 FT-K
	AXIAL FORCE =	-16.718	KIPS		

C. Sample Problem 2
(CASE B)

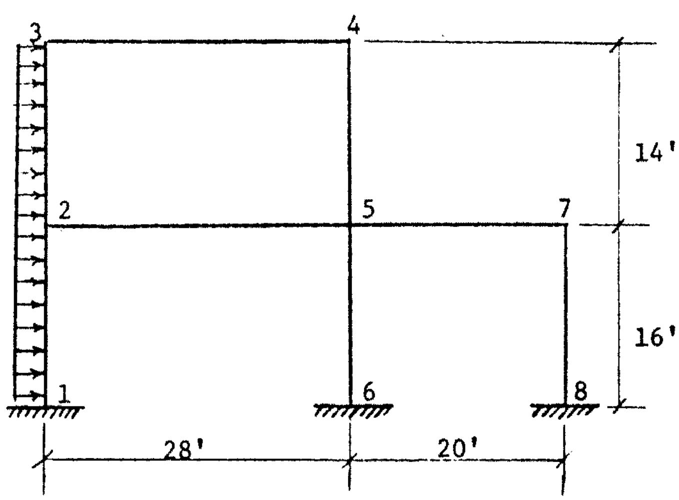


TABLE 3.7-21B (Cont'd)

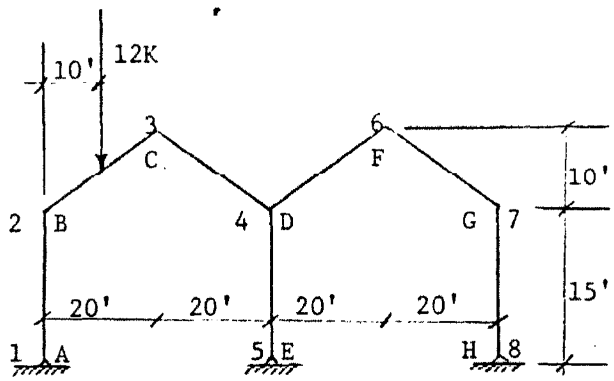
		RFRM RESULTS			REFERENCE RESULTS
FOR THE MEMBER	3 4				
	FX(3) =	6.38496	KIPS		
	FY(3) =	-2.43141	KIPS		
	M (3) =	28.26689	FT-K		M(3) = 28.4 FT-K
	FX(4) =	-6.38496	KIPS		
	FY(4) =	2.43141	KIPS		
	M (4) =	39.81273	FT-K		M(4) = 39.8 FT-K
	AXIAL FORCE =	-6.385	KIPS		
FOR THE MEMBER	2 5				
	FX(2) =	21.74075	KIPS		
	FY(2) =	-9.70459	KIPS		
	M (2) =	147.89689	FT-K		M(2) = 147.17 FT-K
	FX(5) =	-21.74075	KIPS		
	FY(5) =	9.70459	KIPS		
	M (5) =	123.83150	FT-K		M(5) = 124.18 FT-K
	AXIAL FORCE =	-21.741	KIPS		
FOR THE MEMBER	5 7				
	FX(5) =	10.35020	KIPS		
	FY(5) =	-6.54840	KIPS		
	M (5) =	59.19128	FT-K		M(5) = 59.14 FT-K
	FX(7) =	-10.35020	KIPS		
	FY(7) =	6.54840	KIPS		
	M (7) =	71.77667	FT-K		M(7) = 72.76 FT-K
	AXIAL FORCE =	-10.350	KIPS		
FOR THE MEMBER	3 2				
	FX(3) =	-6.38496	KIPS		
	FY(3) =	2.43141	KIPS		
	M (3) =	-28.26689	FT-K		M(3) = -28.42 FT-K
	FX(2) =	-21.61504	KIPS		
	FY(2) =	-2.43141	KIPS		
	M (2) =	-78.34363	FT-K		M(2) = -78.62 FT-K
	AXIAL FORCE =	2.431	KIPS		

TABLE 3.7-21B (Cont'd)

RFRM RESULTS					REFERENCE RESULTS
FOR THE MEMBER	2 1				
	FX(2) =	-0.12571	KIPS		
	FY(2) =	12.13600	KIPS		
	M (2) =	-69.55325	FT-K		M(2) =-68.55 FT-K
	FX(1) =	-31.87429	KIPS		
	FY(1) =	-12.13600	KIPS		
	M (1) =	-184.43536	FT-K		M(1) =-182.24 FT -K
	AXIAL FORCE =	12.136	KIPS		
FOR THE MEMBER	4 5				
	FX(4) =	6.33496	KIPS		
	FY(4) =	-2.43141	KIPS		
	M (4) =	-39.81273	FT-K		M(4) = -39.84 FT-K
	FX(5) =	-6.38496	KIPS		
	FY(5) =	2.43141	KIPS		
	M (5) =	-49.57675	FT-K		M(5) = -49.12 FT-K
	AXIAL FORCE =	-2.431	KIPS		
FOR THE MEMBER	5 6				
	FX(5) =	17.77551	KIPS		
	FY(5) =	-5.58760	KIPS		
	M (5) =	-133.44603	FT-K		M(5)=-134.30 FT- K
	FX(6) =	-17.77551	KIPS		
	FY(6) =	5.58760	KIPS		
	M (6) =	-150.96212	FT-K		M(6)=-151.07 FT- K
	AXIAL FORCE =	-5.588	KIPS		
FOR THE MEMBER	7 8				
	FX(7) =	10.35020	KIPS		
	FY(7) =	-6.54840	KIPS		
	M (7) =	-71.77667	FT-K		M(7)= -72.76 FT- K
	FX(8) =	-10.35020	KIPS		
	FY(8) =	6.54840	KIPS		
	M (8) =	-93.82657	FT-K		M(8)= -95.16 FT- K
	AXIAL FORCE =	-6.548	KIPS		

TABLE 3.7-21B (Cont'd)

D. Sample Problem No. 3: Reference - Moment Distribution
J.M. Gere
Page 102 & 366



moment sign convention

		RFRM RESULTS			REFERENCE RESULTS
FOR THE MEMBER	2 1				
	FX(2) =	-2.41838	KIPS		
	FY(2) =	-9.37467	KIPS		
	M (2) =	36.27563	FT-K		M(2) = 36.5 FT-K
	FX(1) =	2.41838	KIPS		
	FY(1) =	9.37467	KIPS		
	M (1) =	0.00000	FT-K		
	AXIAL FORCE =	-9.375	KIPS		
FOR THE MEMBER	2 3				
	FX(2) =	2.41838	KIPS		
	FY(2) =	9.37467	KIPS		
	M (2) =	-36.27563	FT-K		
	FX(3) =	-2.41838	KIPS		
	FY(3) =	2.62533	KIPS		
	M (3) =	-7.03039	FT-K		
	AXIAL FORCE =	-6.356	KIPS		

TABLE 3.7-21B (Cont'd)

RFRM RESULTS					REFERENCE RESULTS
FOR THE MEMBER	3 4				
	FX(3) =	2.41838	KIPS		
	FY(3) =	-2.62533	KIPS		
	M (3) =	7.03039	FT-K		
	FX(4) =	-2.41838	KIPS		
	FY(4) =	2.62533	KIPS		
	M (4) =	21.25242	FT-K		
	AXIAL FORCE =	-3.337	KIPS		
FOR THE MEMBER	4 5				
	FX(4) =	1.95960	KIPS		
	FY(4) =	-2.25075	KIPS		
	M (4) =	-29.39406	FT-K		M(4) = -29.4 FT-K
	FX(5) =	-1.95960	KIPS		
	FY(5) =	2.25075	KIPS		
	M (5) =	0.00000	FT-K		
	AXIAL FORCE =	-2.251	KIPS		
FOR THE MEMBER	4 6				
	FX(4) =	0.45877	KIPS		
	FY(4) =	-0.37458	KIPS		
	M (4) =	8.10163	FT-K		
	FX(6) =	-0.45877	KIPS		
	FY(6) =	0.37458	KIPS		
	M (6) =	3.97759	FT-K		
	AXIAL FORCE =	-0.243	KIPS		
FOR THE MEMBER	6 7				
	FX(6) =	0.45877	KIPS		
	FY(6) =	-0.37458	KIPS		
	M (6) =	-3.97769	FT-K		
	FX(7) =	-0.45877	KIPS		
	FY(7) =	0.37458	KIPS		
	M (7) =	6.88158	FT-K		
	AXIAL FORCE =	-0.578	KIPS		

TABLE 3.7-21B (Cont'd)

	RFRM RESULTS				REFERENCE RESULTS
FOR THE MEMBER,	7 8				
	FX(7) =	0.45877	KIPS		
	FY(7) =	-0.37458	KIPS		
	M (7) =	-6.88158	FT-K		M(7) = -7.0 FT-K
	FX(8) =	-0.45877	KIPS		
	FY(8) =	0.37485	KIPS		
	M (8) =	-0.00000	FT-K		
	AXIAL FORCE =	-0.375	KIPS		

TABLE 3.7-22

NATURAL FREQUENCIES-AND DOMINANT DEGREES OF FREEDOM
(Historical Data Only) SHEET 1

Mode Number	Frequency (cps)	Names	Dominant Degrees of Freedom Direction	Locations
1	3.32	RI1	Z	Reactor Internals
2	3.32	RI1	X	Reactor Internals
3	4.67	M61	X, Y, Z	Pump 1A2
4	4.69	M52	X, Y, Z	Pump 1B1
5	5.09	M66	X, Y	Pump 1A1
6	5.11	M43	X, Y	Pump 1B2
7	8.01	M66	Z, X	Pump 1A1
8	8.03	M43	Z, X	Pump 1B2
9	8.87	M61	Z, X	Pump 1A2
10	8.89	M52	Z, X	Pump 1B1
11	10.51	SG5A & B	X	Steam Generators 1A & 1B
12	10.51	SG5A & B	X	Steam Generators 1A & 1B
13	10.58	M61	X, Y	Pump 1A2
14	10.70	M52	X, Y	Pump 1B1
15	11.01	M66	X, Y, Z	Pump 1A1
16	11.12	M43	X, Y, Z	Pump 1B2
17	12.14	RI2	Z	Reactor Internals
18	12.16	RI2	X	Reactor Internals
19	19.84	SG9A & B	Z	Steam Generators 1A & 1B
20	19.86	SG9A & B	Z	Steam Generators IA & 1B
21	23.06	M65	Z	Pump 1A1
22	23.10	M42	Z	Pump 1B2
23	23.65	RI1	y	Reactor Internals
24	24.52	M60	Z	Pump 1A2
25	24.56	M51	Z	Pump 1B1
26	27.39	SG9A & B	X	Steam Generators 1A & 1B
27	27.68	SG5A & B	y	Steam Generators 1A & 1B
28	27.80	SG5A & B	y	Steam Generators 1A & 1B
29	30.27	SG9A & B	X	Steam Generators IA & 1B
30	37.19	V1	Z	Reactor Vessel
31	39.22	SG5A & B	Z	Steam Generators 1A & 1B
32	39.24	SG5A & B	Z	Steam Generators 1A & 1B
33	39.71	M60	X	Pump 1A2
34	41.01	M51	X	Pump 1B1
35	42.25	M65	X	Pump 1A1
36	42.84	M42	X	Pump 1B2
37	44.96	V1	X	Reactor Vessel
38	49.90	V1	Z	Reactor Vessel
39	50.75	V4	X	Reactor Vessel

Note:
The steam generator modes (Modes 11, 12, 19, 20, 26, 27, 28, 29, 31, and 32 of Sheet 1) have been evaluated for a conservative decrease of two percent in frequency to account for the additional mass of the replacement steam generator. All frequency changes for other components are insignificant.

TABLE 3.7-22 (Cont'd)

SHEET 2

Mode Number	Frequency (cps)	Dominant Degrees of Freedom		
		Names	Direction	Locations
1	14.34	M5	X	Pressurizer
2	61.14	M5	Y	"
3	62.60	M5	X	"
4	167.82	M5	X	"
5	209.94	M1	X	"
6	363.32	M2	X	"
1	5.65	M5	Y	Surge Line
2	13.82	H1	X, Z	" "
3	16.42	M6, H3	X	" "
4	20.94	H1	Y	" "
5	24.48	H3, M7	Z	" "
6	30.65	M6, M4	Y	" "
7	38.23	M7, M8	X	" "
8	49.21	M4, M5	X	" "
9	69.70	M3	X, Y	" "
10	74.15	H3	Y	" "
11	84.56	H3, M8	X	" "
12	157.63	M5, M4	X, Z	" "
13	162.21	M5, M4	Y	" "
14	179.83	M3	Y	" "
15	206.49	M7	Y	" "
16	230.90	M7	X	" "
17	254.34	M7	Y	" "
18	293.40	M7	Y	" "
19	348.27	M8, M7	Y	" "
20	403.80	M4, M8, M5	X, Y, Z	" "
21	515.11	M6, M4	Z	" "
22	623.61	M8, M7	Y	" "
23	648.56	M2	X, Z	" "
24	685.94	M5, M4	Z	" "
25	785.87	M2	X, Z	" "
26	883.67	M7	Z	" "
27	1831.5	M2	Y	" "

TABLE 3.7-23
SEISMIC LOADS ON REACTOR COOLANT SYSTEM
COMPONENTS FOR OPERATIONAL BASIS EARTHQUAKE

(Historical Data Only)

Seismic Excitation	Component and Design Location	Component	Seismic Load		
			Original Calculated Maximum	RSG Maximum	Specified for Design
Combined X and Y	Reactor Vessel Outlet Nozzle and Reactor Vessel Outlet Piping	Fx (kips)	60.4	62.2	241.0
		Fy (kips)	15.2	15.7	253.0
		Fz (kips)	1.0	1.0	10.0
		Mx (in-kips)	15.8	16.3	1042.0
		My (in-kips)	95.3	98.2	1041.0
		Mz (in-kips)	1267.5	1,305.5	43733.0
	Reactor Vessel Inlet Nozzle	Fx (kips)	50.8	52.3	190.0
		Fy (kips)	17.9	18.4	146.0
		Fz (kips)	47.4	48.8	57.0
		Mx (in-kips)	2262.8	2,330.7	9638.0
		My (in-kips)	1517.0	1,562.5	5981.0
		Mz (in-kips)	1876.8	1,933.1	12770.0
Combined Z and Y	Reactor Vessel Outlet Nozzle and Reactor Vessel Outlet Piping	Fx (kips)	16.9	17.4	30.0
		Fy (kips)	2.4	2.5	172.0
		Fz (kips)	6.4	6.6	40.0
		Mx (in-kips)	150.0	154.5	1170.0
		My (in-kips)	800.0	824.0	7521.0
		Mz (in-kips)	244.4	251.7	37446.0
	Reactor Vessel Inlet Nozzle	Fx (kips)	29.7	30.6	61.0
		Fy (kips)	12.4	12.8	105.0
		Fz (kips)	29.8	30.7	108.0
		Mx (in-kips)	1874.8	1,931.0	22392.0
		My (in-kips)	731.1	753.0	10085.0
		Mz (in-kips)	1548.9	1,595.4	14087.0

Note: All original calculated maximum loads have been evaluated for a conservative increase of three percent, except for the steam generator upper support which has been evaluated for a conservative increase of eight percent, to account for the increase in mass and center of gravity of the replacement steam generator. These conservatively increased loads are given in the RSG Maximum column of this table. These values show that the seismic loads remained below the bounding specified for design loads with the replacement steam generators.

TABLE 3.7-23 (Cont'd)
SHEET 2

Seismic Excitation	Component and Design Location	Component	Seismic Load		
			Original Calculated Maximum	RSG Maximum	Specified for Design
Combined X and Y	Steam Generator Inlet Nozzle and Steam Generator Inlet Piping	Fx (kips)	60.9	62.7	149.0
		Fy (kips)	52.7	54.3	68.0
		Fz (kips)	1.0	1.0	4.0
		Mx (in-kips)	73.8	76.0	743.0
		My (in-kips)	85.8	88.4	708.0
		Mz (in-kips)	695.8	716.7	16828.0
	Steam Generator Outlet Nozzle	Fx (kips)	26.4	27.2	30.0
		Fy (kips)	31.7	32.7	134.0
		Fz (kips)	26.4	27.2	33.0
		Mx (in-kips)	3018.3	3,108.8	12955.0
		My (in-kips)	2708.2	2,789.4	3161.0
		Mz (in-kips)	3018.3	3,108.8	6686.0
Combined Z and Y	Steam Generator Inlet Nozzle and Steam Generator Inlet Piping	Fx (kips)	17.0	17.5	27.0
		Fy (kips)	15.2	15.7	149.0
		Fz (kips)	6.4	6.6	7.0
		Mx (in-kips)	356.5	367.2	575.0
		My (in-kips)	381.6	393.0	3793.0
		Mz (in-kips)	63.4	65.3	11206.0
	Steam Generator Outlet Nozzle	Fx (kips)	27.8	28.6	30.0
		Fy (kips)	22.0	22.7	93.0
		Fz (kips)	27.8	28.6	119.0
		Mx (in-kips)	1866.0	1,922.0	6467.0
		My (in-kips)	1586.4	1,634.0	4251.0
		Mz (in-kips)	1866.0	1,922.0	7140.0

TABLE 3.7-23 (Cont'd)

SHEET 3

Seismic Excitation	Component and Design Location	Component	Seismic Load		
			Original Calculated Maximum	RSG Maximum	Specified for Design
Combined X and Y	Reactor Coolant Pump Inlet Nozzle	Fx (kips)	10.7	11.0	78.2
		Fy (kips)	31.7	32.7	137.0
		Fz (kips)	4.5	4.6	37.9
		Mx (in-kips)	1181.1	1,216.5	5713.2
		My (in-kips)	707.6	728.8	4976.0
		Mz (in-kips)	3357.1	3,457.8	16461.0
	Reactor Coolant Pump Outlet Nozzle	Fx (kips)	51.0	52.5	199.3
		Fy (kips)	17.2	17.7	145.9
		Fz (kips)	37.8	38.9	40.0
		Mx (in-kips)	1347.2	1,387.6	3989.0
		My (in kips)	2180.3	2,245.7	8583.9
		Mz (in-kips)	2300.8	2,369.8	16658.0
Combined Z and Y	Reactor Coolant Pump Inlet Nozzle	Fx (kips)	15.3	15.8	39.1
		Fy (kips)	21.9	22.6	94.9
		Fz (kips)	11.6	11.9	109.9
		Mx (in-kips)	938.6	966.8	11173.0
		My (in-kips)	768.0	791.0	11862.0
		Mz (in-kips)	2451.3	2,524.8	8334.1
	Reactor Coolant Pump Outlet Nozzle	Fx (kips)	29.8	30.7	125.1
		Fy (kips)	12.4	12.8	91.4
		Fz (kips)	19.7	20.3	94.9
		Mx (in-kips)	839.2	864.4	17532.0
		My (in-kips)	1031.1	1,062.0	1100.0
		Mz (in-kips)	1262.0	1,299.9	6391.0

TABLE 3.7-23 (Cont'd)

Sheet 4

Seismic Excitation	Component and Design Location	Component	Seismic Load		
			Original Calculated Maximum	RSG Maximum	Specified for Design
Combined X and Y	Steam Generator Outlet Piping	M (in-kips)	3323.6	3,423.3	12000.0
	Steam Generator Outlet Piping	M (in-kips)	2235.1	2,302.2	12000.0
	Pump Inlet Piping	M (in-kips)	1794.0	1,847.8	12000.0
	Pump Inlet Piping	M (in-kips)	3628.1	3,736.9	12000.0
	Pump Outlet Piping	M (in-kips)	2955.1	3,043.8	12000.0
	R.V. Inlet Piping	M (in-kips)	2636.1	2,715.2	12000.0
Combined Z and Y	Steam Generator Outlet Piping	M (in-kips)	1974.1	2,033.3	12000.0
	Steam Generator Outlet Piping	M (in-kips)	1537.8	1,583.9	12000.0
	Pump Inlet Piping	M (in-kips)	1193.5	1,229.3	12000.0
	Pump Inlet Piping	M (in-kips)	2627.3	2,706.1	12000.0
	Pump Outlet Piping	M (in-kips)	1318.2	1,357.7	12000.0
	R.V. Inlet Piping	M (in-kips)	1952.8	2,011.4	12000.0

TABLE 3.7-23 (Cont'd)
Sheet 5

Seismic Excitation	Component and Design Location	Component	Seismic Load		
			Original Calculated Maximum	RSG Maximum	Specified for Design
Combined X and Y	Reactor Vessel Outlet Support	H (kips)	5.9	6.1	22.0
		V (kips)	214.7	221.1	473.0
	Reactor Vessel Inlet Support	H (kips)	261.3	269.1	1052.0
		V (kips)	122.7	126.4	469.0
	Steam Generator Lower Support	Fy (kips)	187.5	193.1	624.0
		Fz (kips)	44.8	46.1	54.0
		Mx (in-kips)	0.1	0.1	21.0
		My (in-kips)	156.5	161.2	455.0
		Mz (in-kips)	2700.2	2,781.2	24383.0
	Steam Generator Upper Support	Fx (kips)	123.4	133.3	140.0
	Pressurizer Support	Fx (kips)	22.2	22.9	82.5
		Fy (kips)	24.6	25.3	80.7
		Mz (in-kips)	5681.8	5,852.3	17207.4
	Reactor Coolant Pump Vertical Support	Fy (kips)	1.1	1.1	4.6
	Reactor Coolant Pump Horiz. Support	Fa (kips)	5.6	5.8	25.0

TABLE 3.7-23 (Cont'd)
Sheet 6

Seismic Excitation	Component and Design Location	Component	Seismic Load		
			Original Calculated Maximum	RSG Maximum	Specified for Design
Combined Z and Y	Reactor Vessel Outlet Support	H (kips)	257.5	265.2	663.0
		V (kips)	37.7	38.8	392.0
	Reactor Vessel Inlet Support	H (kips)	134.4	138.4	304.0
		V (kips)	177.3	182.6	692.0
	Steam Generator Lower Support	Fy (kips)	172.9	178.1	405.0
		Fz (kips)	81.6	84.0	397.0
		Mx (in-kips)	5019.2	5,169.8	24422.0
		My (in-kips)	722.4	744.1	9772.0
		Mz (in-kips)	0.3	0.3	4132.0
	Steam Generator Upper Support	Fz (kips)	64.6	69.8	240.0
	Pressurizer Support	Fx (kips)	24.6	25.3	80.6
		Fy (kips)	22.2	22.9	82.9
		Mx (in-kips)	5681.8	5,852.3	17101.5
	Reactor Coolant Pump Vertical Support	Fy (kips)	0.7	0.7	9.2
	Reactor Coolant Pump Horiz. Support	Fa (kips)	18.9	19.5	25.0

THIS PAGE INTENTIALLY LEFT BLANK

TABLE 3.7-24

REACTOR INTERNALS AND CORE LOAD AND STRESS CRITERIA

PLANT CONDITION	LOAD COMBINATION	APPLICABLE COMPONENT	STRESS LIMITS
Upset	Normal Operating + Operating Basis Earthquake	Internals	Figure NG 3221.1 including notes
		Core	$P_m \leq S_m$
			$P_B + P_M \leq 1.5\ S_m$
Emergency	Normal Operating + Design Basis Earthquake	Internals	Figure NG 3224.1 including notes
		Core	$P_m \leq 1.5\ S_m$
			$P_B + P_m \leq 2.25\ S_m$
Faulted	Normal Operating + Safe Design Basis + Loss of Coolant Accident	Internals	Appendix F Rules for Evaluating Faulted Conditions

P_m = general primary membrane stress

P_B = primary bending stress

S_m = maxinum allowable stress as defined by the ASME Code.

The maximum allowable stresses of components composed of materials not covered by the code with the exception of zirconium based alloys, shall be calculated as directed by Section III, ASME Boiler and Pressure Vessel Code for materials of similar properties. The maximum allowable stresses, S_m, of zirconium based alloys shall not exceed two-thirds of the unirradiated minimum yield strength at temperature.

TABLE 3.7-25

NATURAL FREQUENCIES FOR

VERTICAL SEISMIC ANALYSIS MATHEMATICAL MODEL

(Historical Data Only)

Mode No.	Frequency, cps	
	Sub-Model I	Sub-Model II
1	21.60	72.98
2	67.75	404.09
3	124.59	--

TABLE 3.7-26

SEISMIC STRESSES IN CRITICAL REACTOR INTERNALS COMPONENTS
FOR THE DESIGN BASIS EARTHQUAKE

(Historical Data Only)

Structural Component	Location	Stress Mode	Design Load Stress	Dynamic Analysis Stress
Core Support Barrel	Upper Section of Barrel	Tension & Bending	1,129 psi	907 psi
Lower Core Support Structure Grid Beam	Beam Flange	Bending	5,278 psi	686 psi
CEA Shrouds				
Single &	End of Shroud	Tension & Bending	3,548 psi	1,771 psi
Dual			2,762 psi	1,729 psi
Upper Grid Beams	Center of Beam	Bending	1,652 psi	222 psi
Upper Guide Structure Flange	Junction of Flange & Barrel Cylinder	Tension & Bending	2,823 psi	161 psi

TABLE 3.7-27

SUMMARY OF FLOOR RESPONSE SPECTRA DATA

Design Periods of Piping

Structure	Direction	1st Mode Period (Sec)	2nd Mode Period (Sec)	70% of Minimum 2nd Mode Period (Sec)	Design Period (Sec)
Reactor	Hor.	0.76	0.29	0.20	0.20
Building	Vert.	-	0.50	-	0.20
Reactor	Hor. E-W	0.45	0.24	-	0.15
Auxiliary	Hor. N-S	0.76	0.22	0.15	0.15
Building	Vert. El.82.0	0.32	0.25	-	0.15
	Vert.-Others	0.57	0.33	-	0.15

Design Accelerations Based on Design Periods (0.20 & 0.15 Resp.)

Structure	Direction	EL.	Max. Acc.* (g)	1.5 x Max. Acc.** (g)	Design Acc. For OBE	Design Acc. For DBE
Reactor	Hor.	68.5	0.26	0.39	0.39	0.78
	Hor.	60.0	0.25	0.37	0.37	0.74
Building	Hor.	44.0	0.27	0.40	0.40	0.80
	Hor.	24.0	0.28	0.43	0.43	0.86
	Hor.	18.0	0.32	0.49	0.49	0.98
	Vert.	All	0.13	0.20	0.20	0.40
Reactor	Hor. E-W	82.0	0.31	0.46	0.46	0.92
Auxiliary	Hor. N-S	82.0	0.27	-	0.46	0.92
	Hor. E-W	62.0	0.23	0.35	0.35	0.70
Building	Hor. N-S	62.0	0.15	-	0.35	0.70
	Hor. E-W	43.0	0.20	0.31	0.31	0.62
	Hor. N-S	43.0	0.13	-	0.31	0.62
	Hor. E-W	19.5	0.21	-	0.35	0.70
	Hor. N-S	19.5	0.23	0.35	0.35	0.70
	Hor. E-W	-0.5	0.23	-	0.52	1.04
	Hor. N-S	-0.5	0.34	0.52	0.52	1.04
	Vert.	82.0	0.50	0.75	0.75	1.50
	Vert.	Others	0.40	0.60	0.60	1.20

*For Periods from 0 to Design Period of Piping
**Selecting the Higher of E-W or N-S Horizontal Acceleration for Each Elevation

TABLE 3.7-28

RESULTS OF SAMPLE PIPING SYSTEMS ANALYSIS

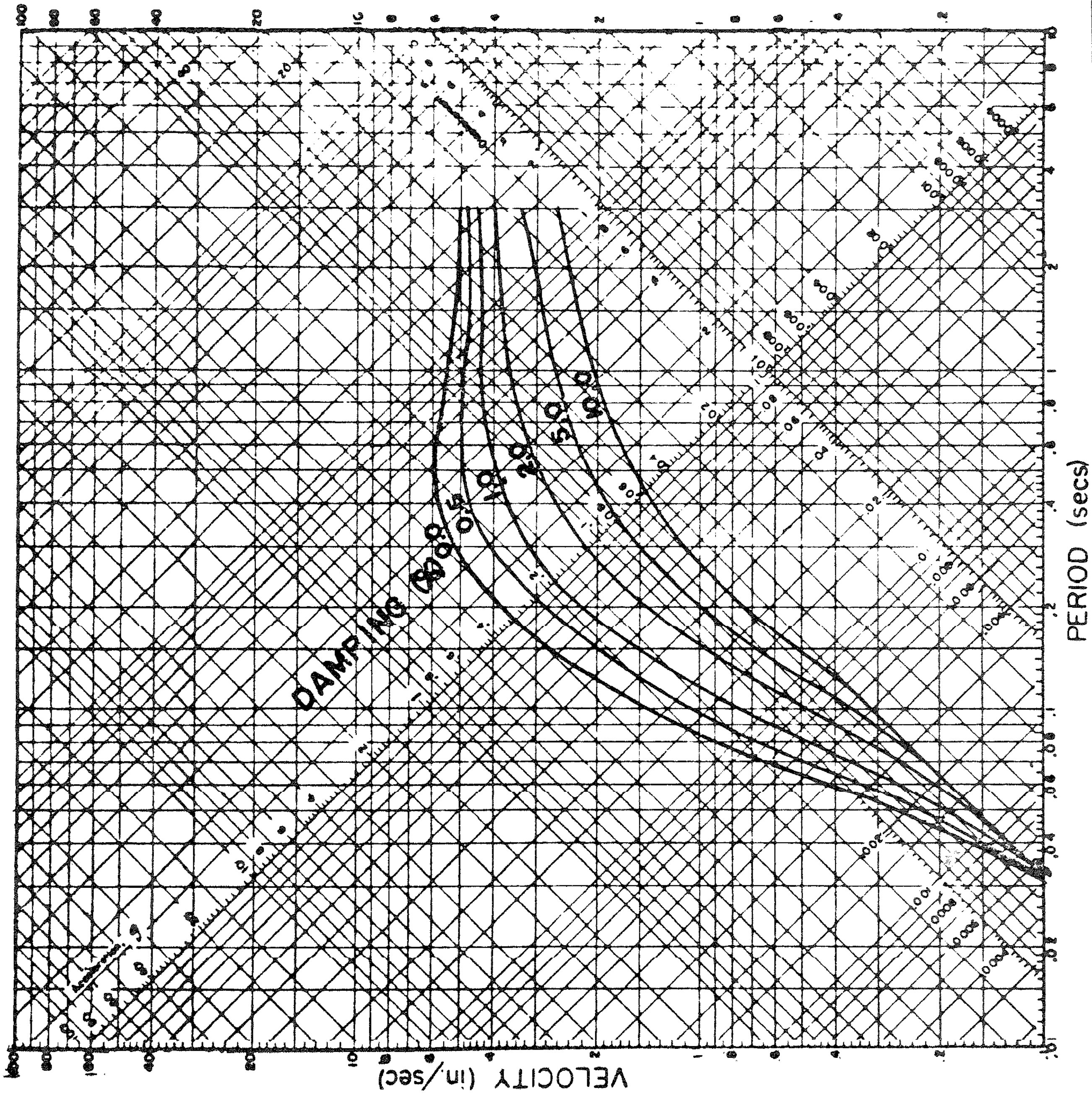
Examples of Locating Restraints for a Preset Design Period

Sample Problem	Pipe Size	Material	Preset Design Period (sec)	Actual Period (sec)
No. 1	2"	Stn. St.	0.20	0.174
No. 2	12"	Stn. St.	0.15	0.142
No. 3	8"	Carb. St.	0.15	0.115

Examples of Conservatism in the 1.5 Participation Factor

			Static analysis with Full Response Loads					Direct Static Analysis			
Sample Problem	Pipe Size	Material	No. of Modes	Period Range (sec)	Acc. Per Mode (g)	Max. Stress (psi)	Dir.	Acc. (g)	Max Stress (psi)	Dir.	Factor of Conservation*
No. 4	8"	Carb. St.	6	0.173 to 0.098	1.0	1639	X-Y	1.5	8372	X-Y	5.11
						292	Y-Z		1968	Y-Z	6.74
No. 5	14"	Carb. St.	5	0.155 to 0.091	1.0	4409	X-Y	1.5	8775	X-Y	1.99
						4403	X-Z		6624	Y-Z	1.50
No. 6	18"	Carb. St.	6	0.195 to 0.085	1.0	2190	X-Y	1.5	5880	X-Y	2.68
						5296	Y-Z		9107	Y-Z	1.72

$$\text{*Factor of Conservatism} = \frac{\text{Max. Stress by Direct Static Analysis}}{\text{Max. Stress by Full Response Analysis}}$$

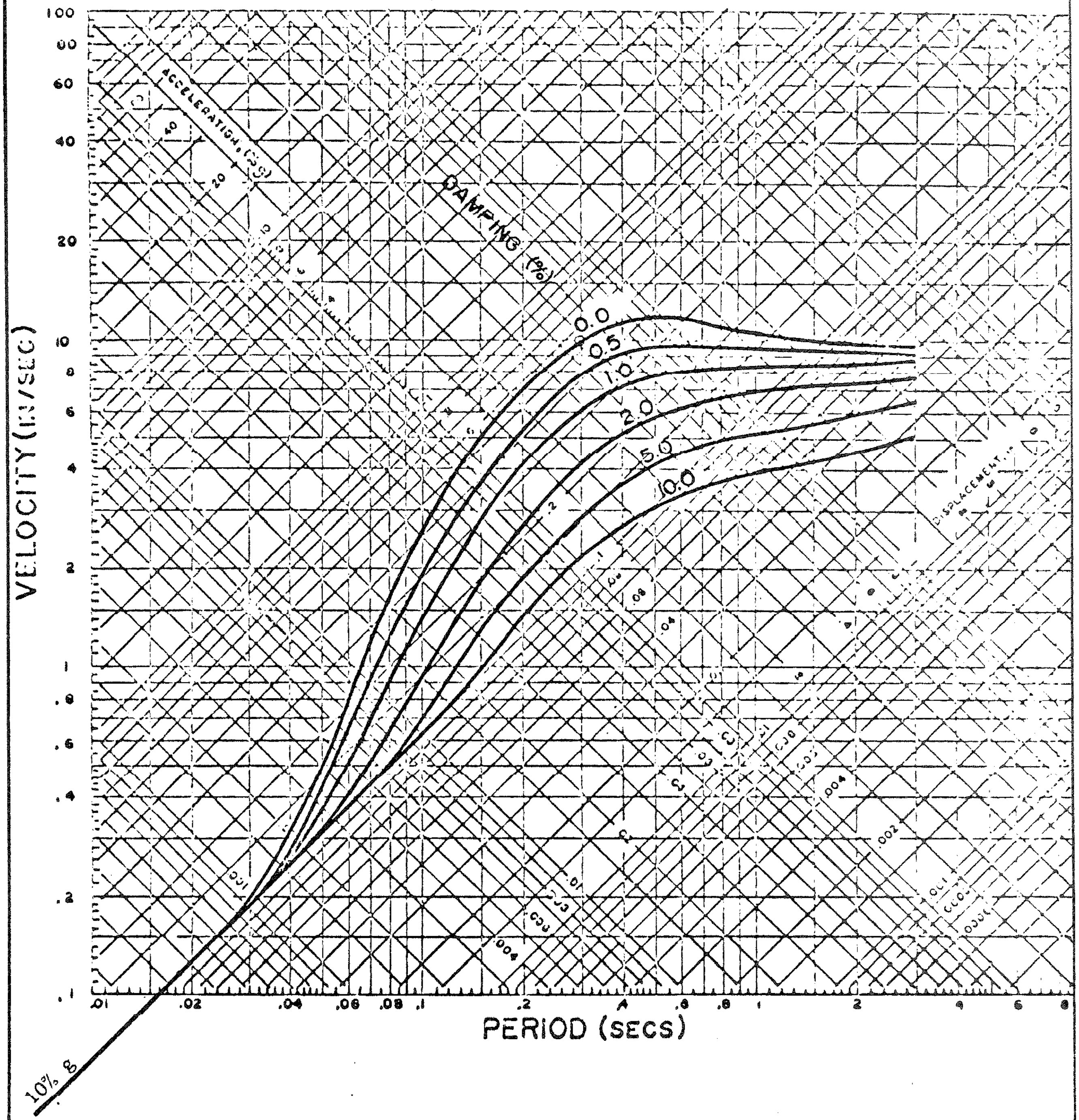


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

OPERATING BASIS EARTHQUAKE
DESIGN RESPONSE (HOUSNER) SPECTRUM

FIGURE 3.7-1



FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

DESIGN BASIS EARTHQUAKE
DESIGN RESPONSE (HOUSNER) SPECTRUM

FIGURE 3.7-2

INPUT PARAMETERS

1. EXPECTED DURATION OF EARTHQUAKE MOTION ASSUMED TO BE 20 SECONDS IN EACH TRIAL.

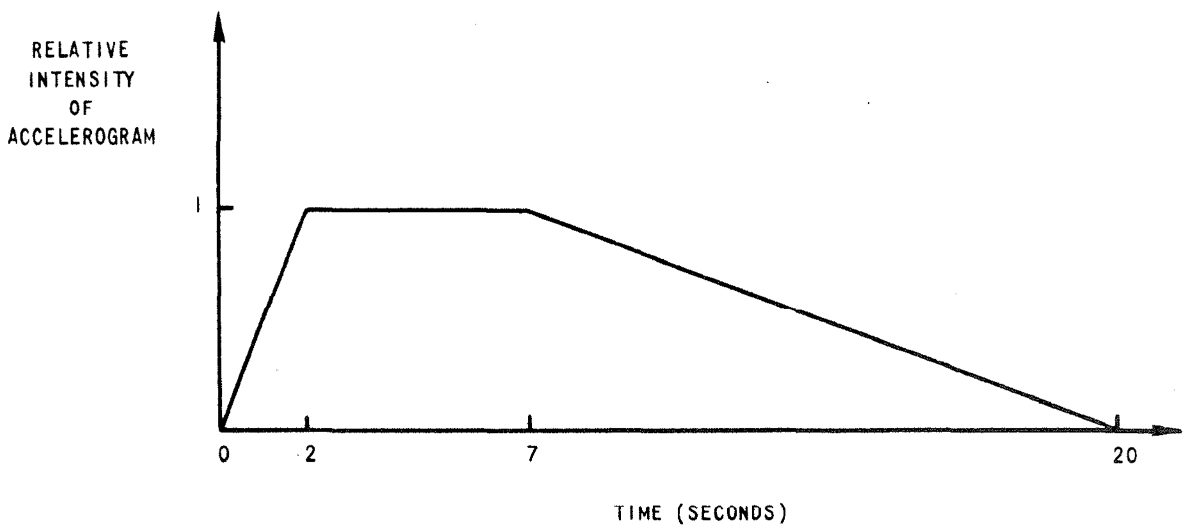
2. EXPECTED MAXIMUM GROUND ACCELERATION RANGED FROM 0.04 g TO 0.10 g DEPENDING UPON THE COMPUTER RUN.

3. SHAPE OF THE POWER SPECTRAL DENSITY FUNCTION:

   TRIAL # 1 WAS BASED ON THE TAJIMI FUNCTION WITH JENNING'S CONSTANTS.

   TRIAL # 2 WAS A SLIGHT MODIFICATION OF THE TAJIMI FUNCTION.

4. SHAPE OF THE INTENSITY FUNCTION WAS THE SAME IN ALL TRIALS.

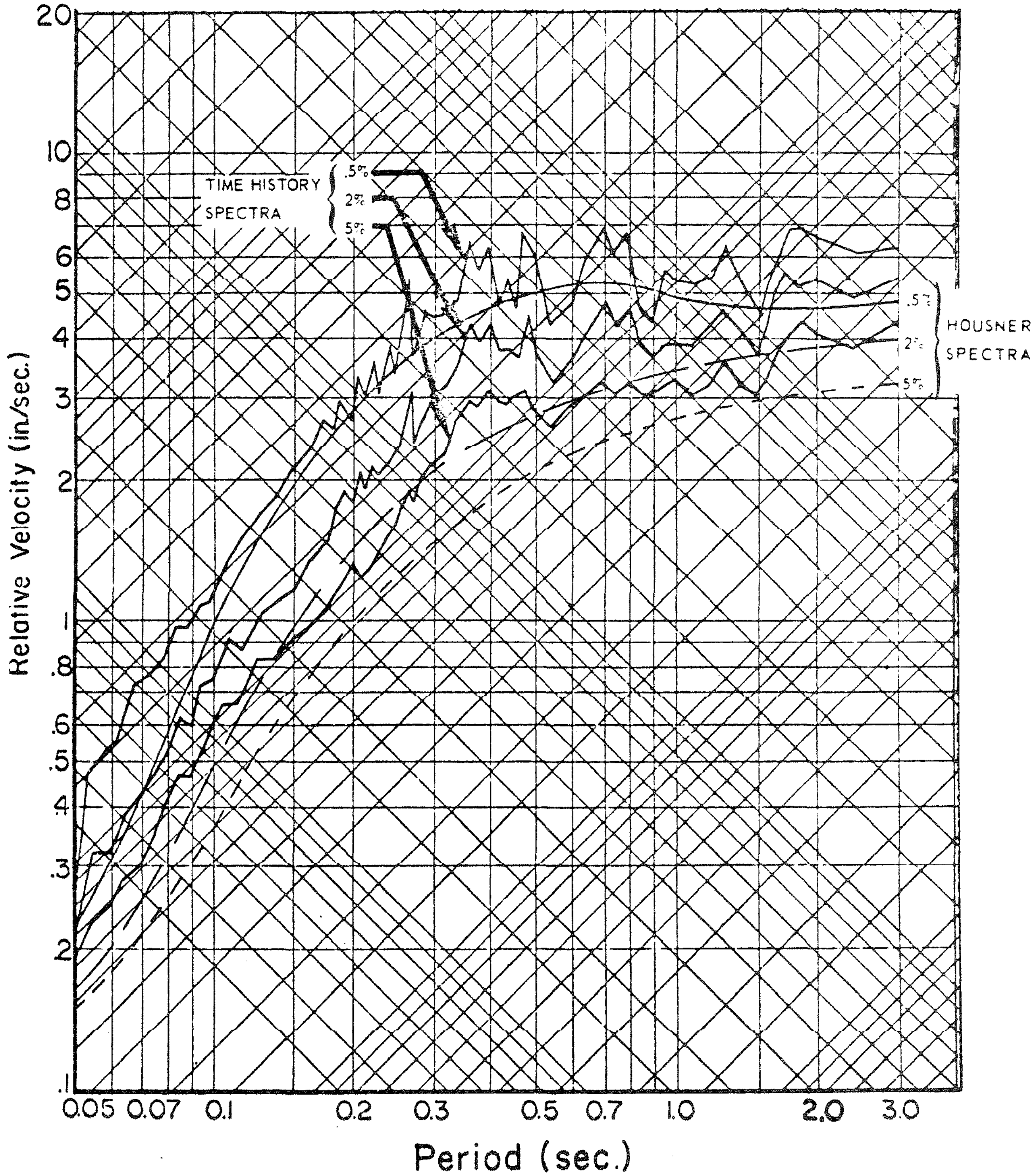


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

SYNTHETIC EARTHQUAKE INTENSITY FUNCTION

FIGURE 3.7-3



NOTE:

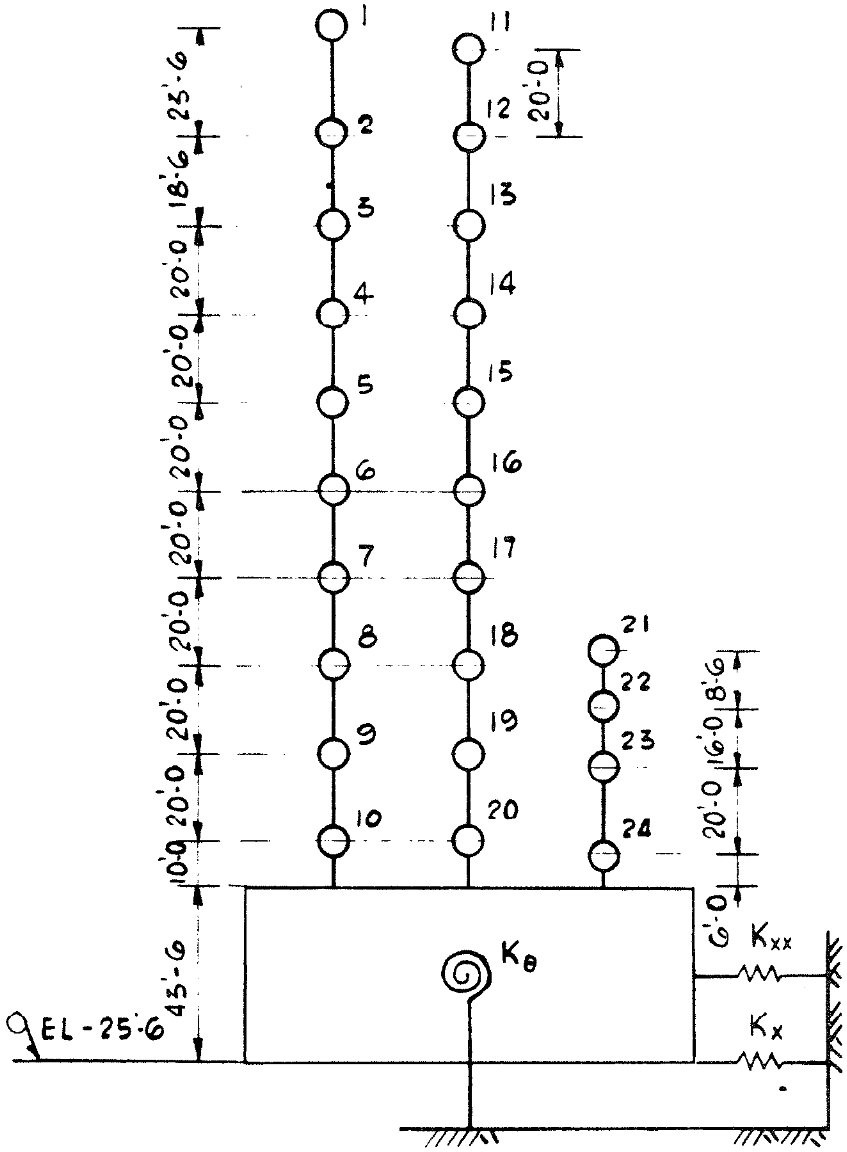
THIS SPECTRA IS FOR OBE.DOUBLE ACCELERATION VALUES FOR DBE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

COMPARISON OF TIME HISTORY AND HOUSNER RESPONSE SPECTRA

FIGURE 3.7-4



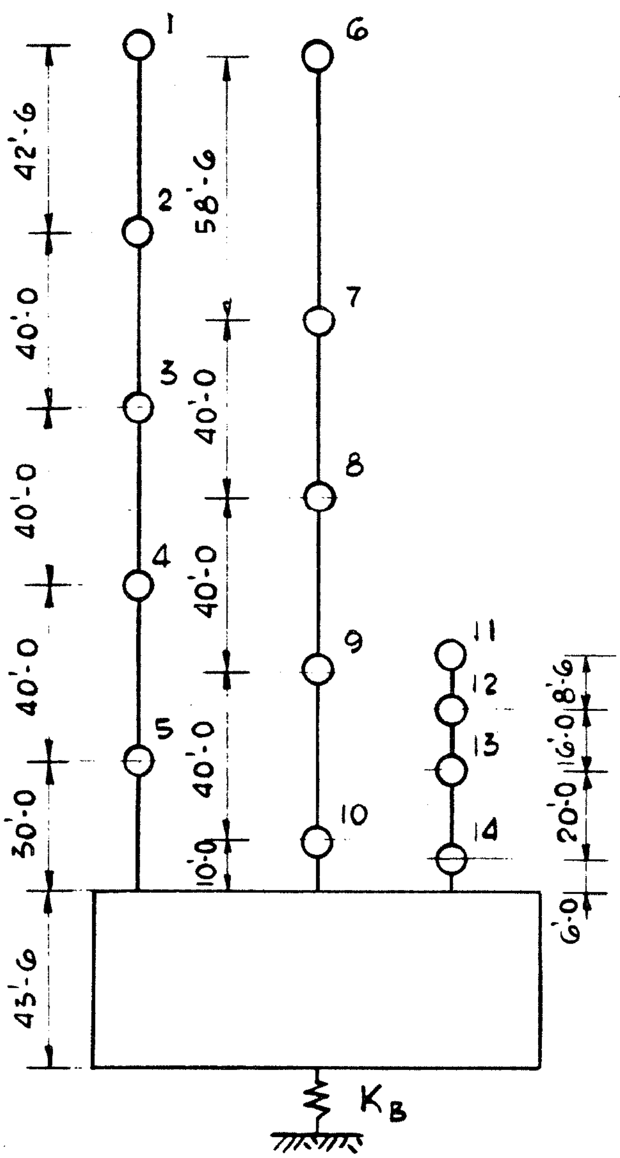
MASS 1 ~ 10 SHIELD STRUCTURE
11 ~ 20 STEEL CONTAINMENT
21 ~ 24 INTERNAL STRUCTURE
DIAMETER OF FDN = 160'-0

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR BUILDING HORIZONTAL
MATHEMATICAL MODEL

FIGURE 3.7-5



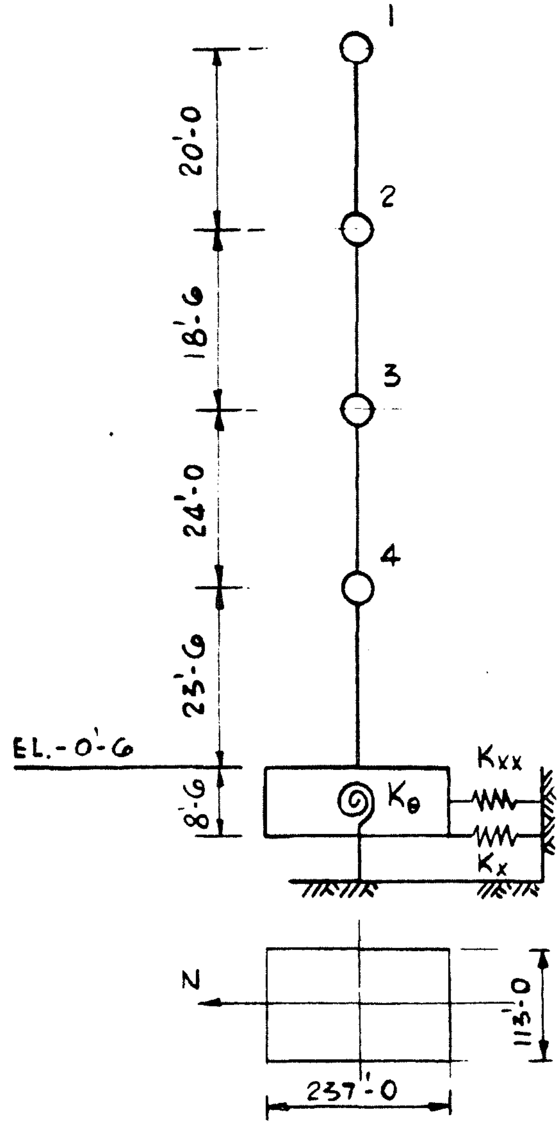
MASS 1 ~ 5 SHIELD STRUCTURE
6 ~ 10 STEEL CONTAINMENT
11 ~ 14 INTERNAL STRUCTURE

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR BUILDING VERTICAL MATHEMATICAL MODEL

FIGURE 3.7-6

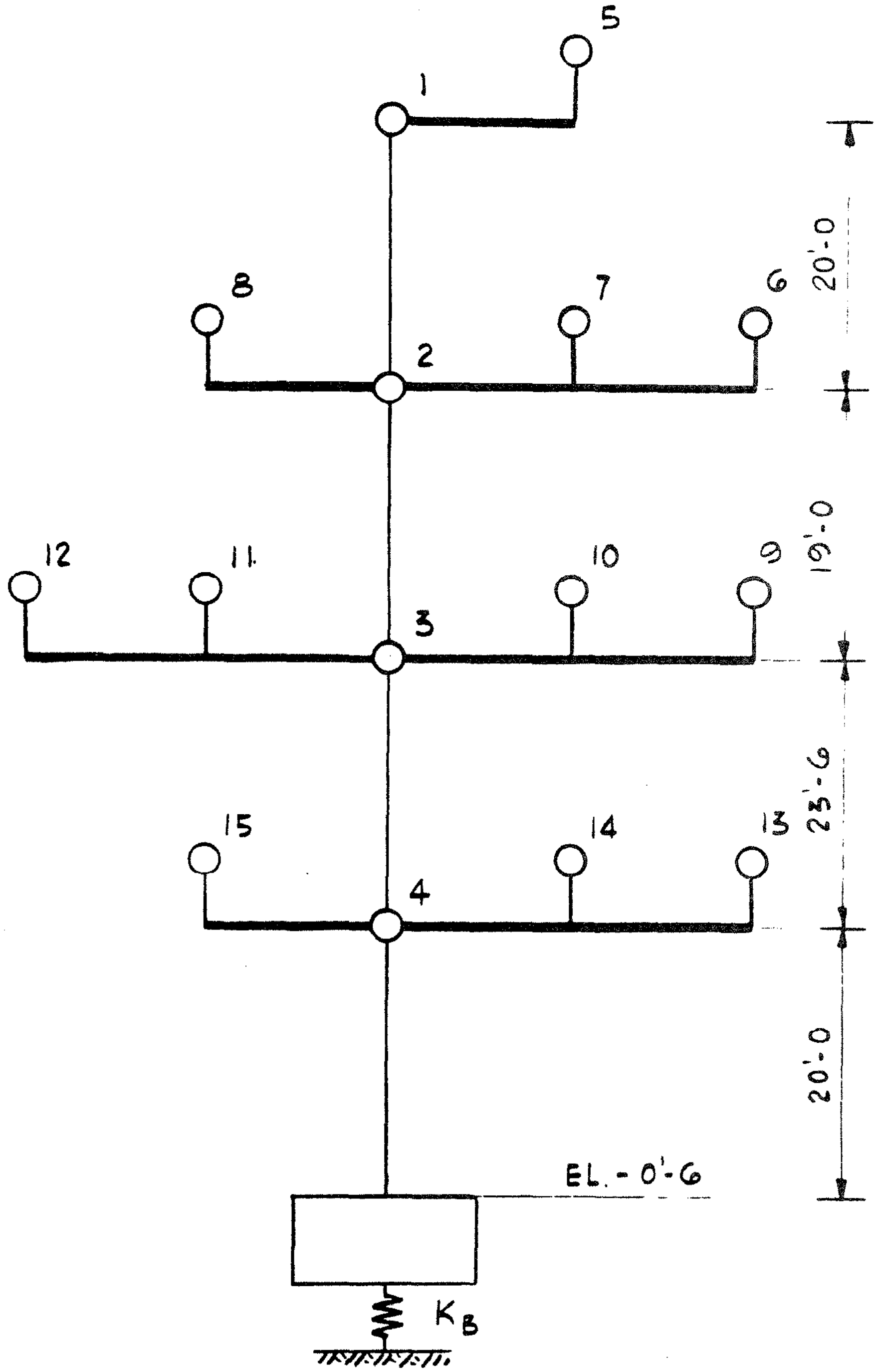


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR AUXILIARY BUILDING
HORIZONTAL MATHEMATICAL MODEL

FIGURE 3.7-7



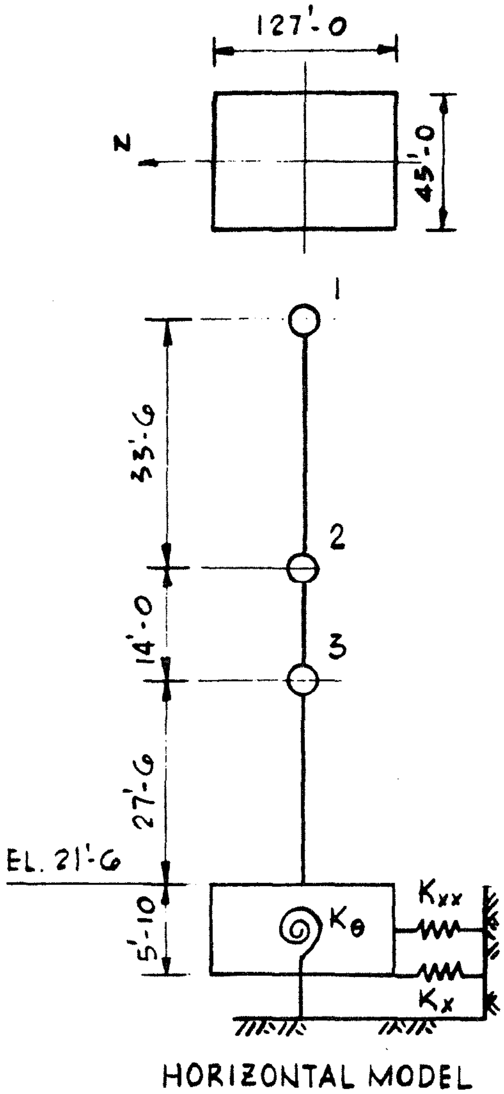
NOTE : HEAVY LINE INDICATES RIGID MEMBER

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR AUXILIARY BUILDING
VERTICAL MATHEMATICAL MODEL

FIGURE 3.7-8

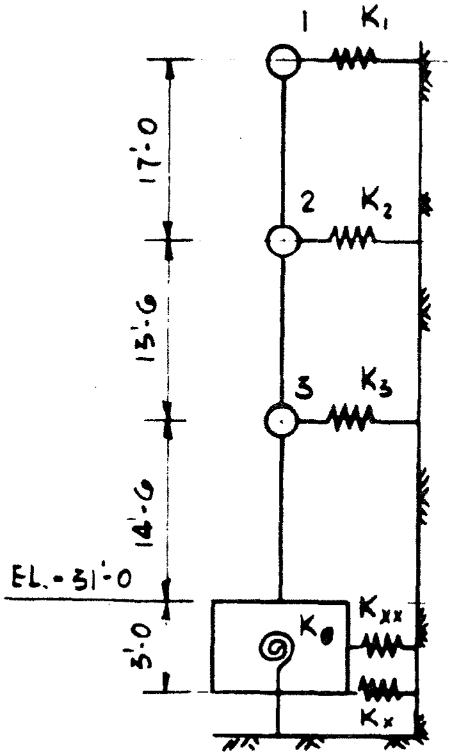


FLORIDA POWER & LIGHT COMPANY

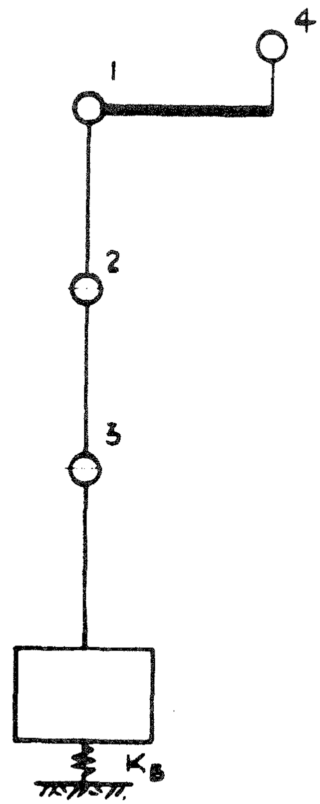
ST. LUCIE PLANT

FUEL HANDLING BUILDING
MATHEMATICAL MODELS

FIGURE 3.7-9



HORIZONTAL MODEL



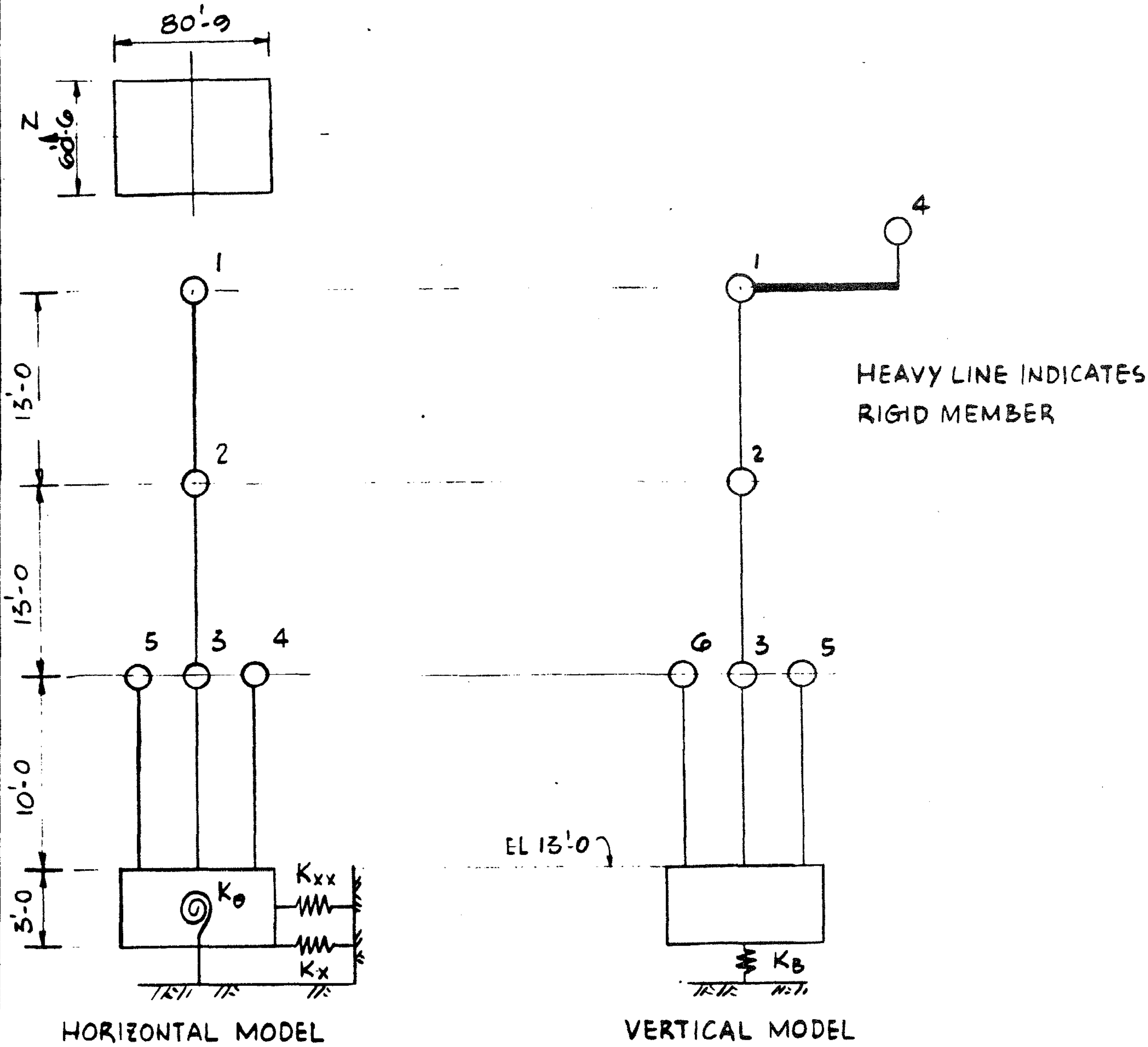
VERTICAL MODEL
HEAVY LINE INDICATES
RIGID MEMBER

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

INTAKE STRUCTURE
MATHEMATICAL MODELS

FIGURE 3.7-10

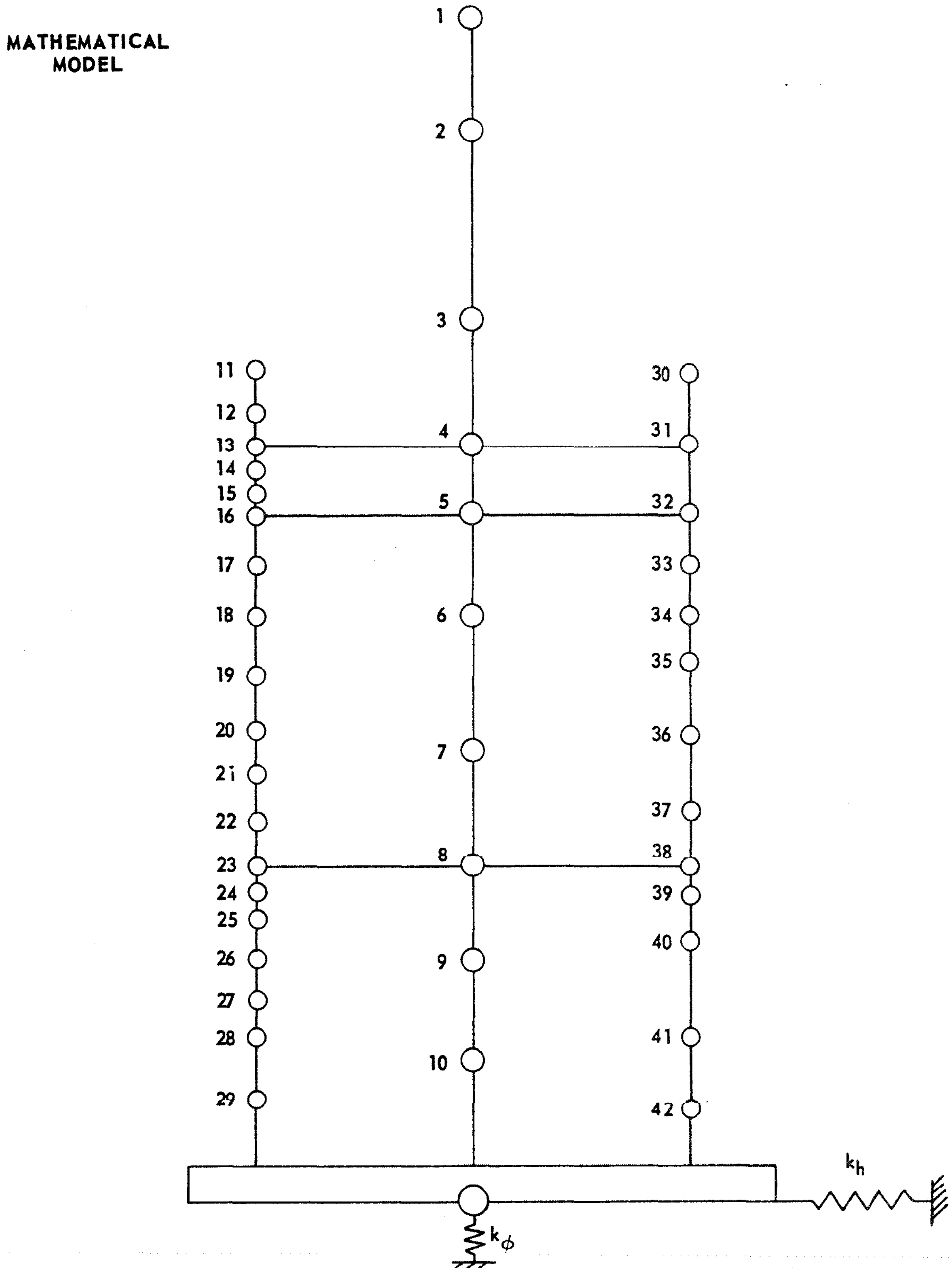


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

DIESEL GENERATOR BUILDING
MATHEMATICAL MODELS

FIGURE 3.7-11

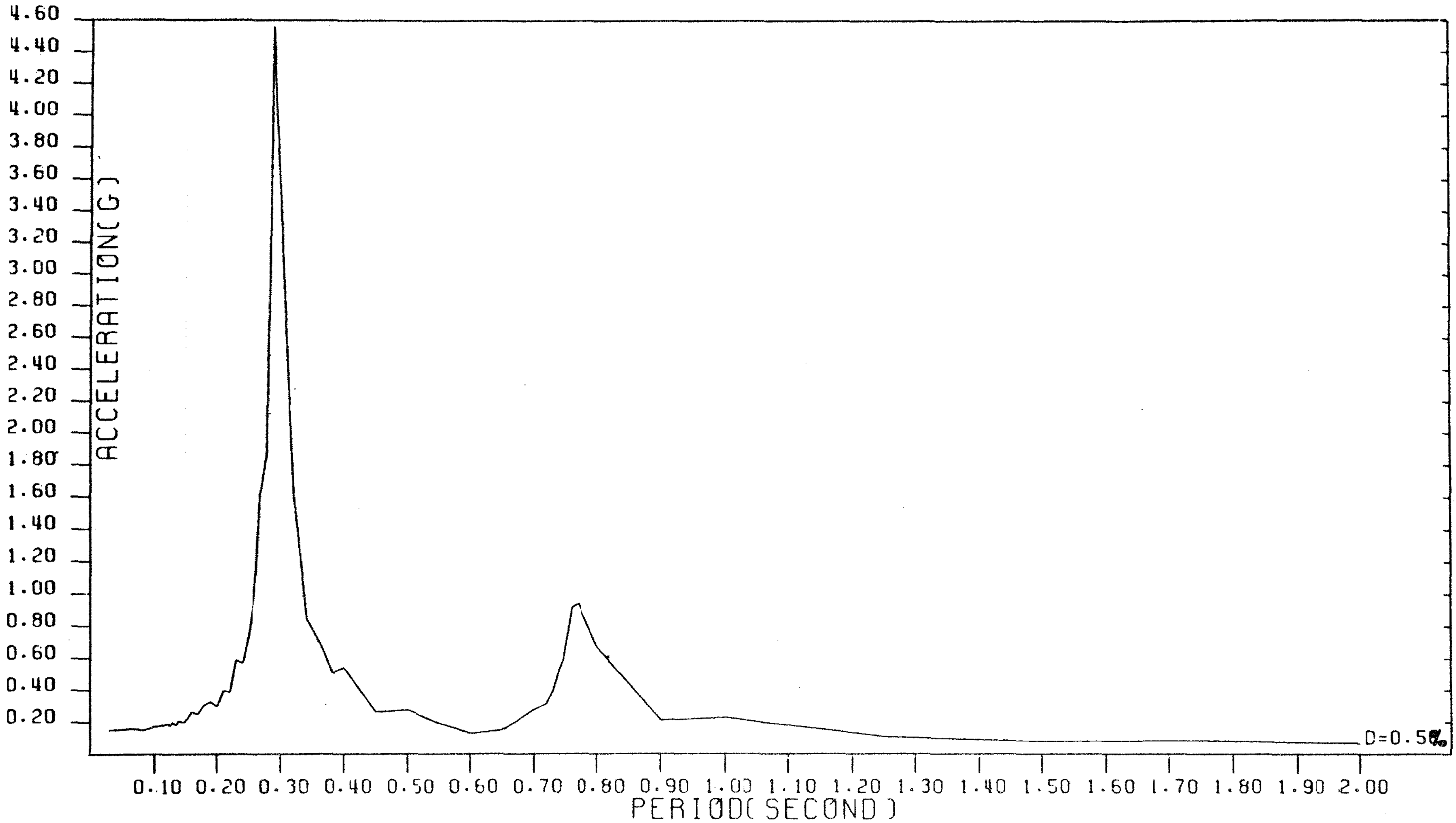


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT UNIT 1

COMPARISON BETWEEN FRQA AND STARDYNE

FIGURE 3.7-11A



NOTE:

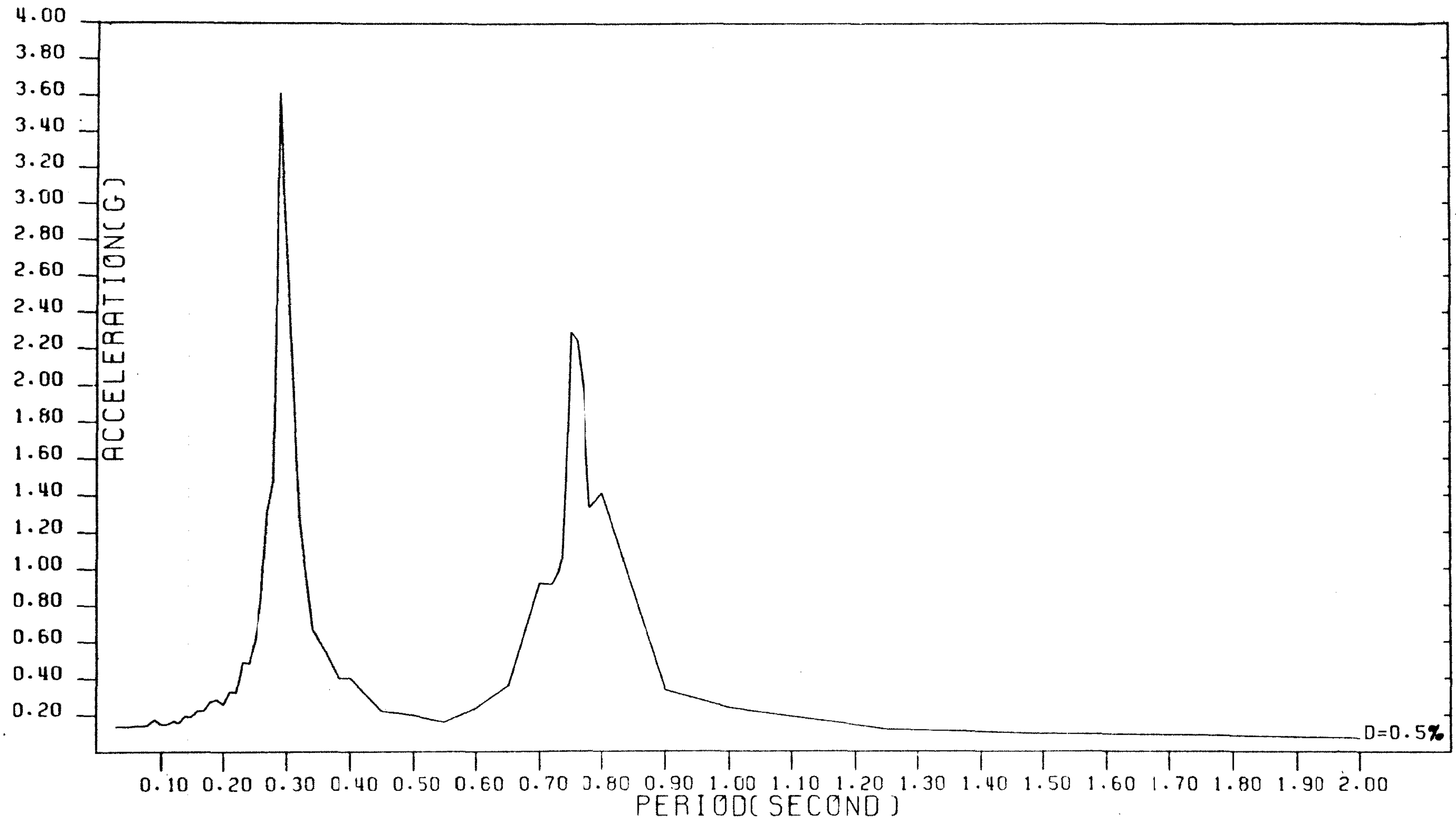
THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR BUILDING EL 18.0
HORIZONTAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-12



NOTE:

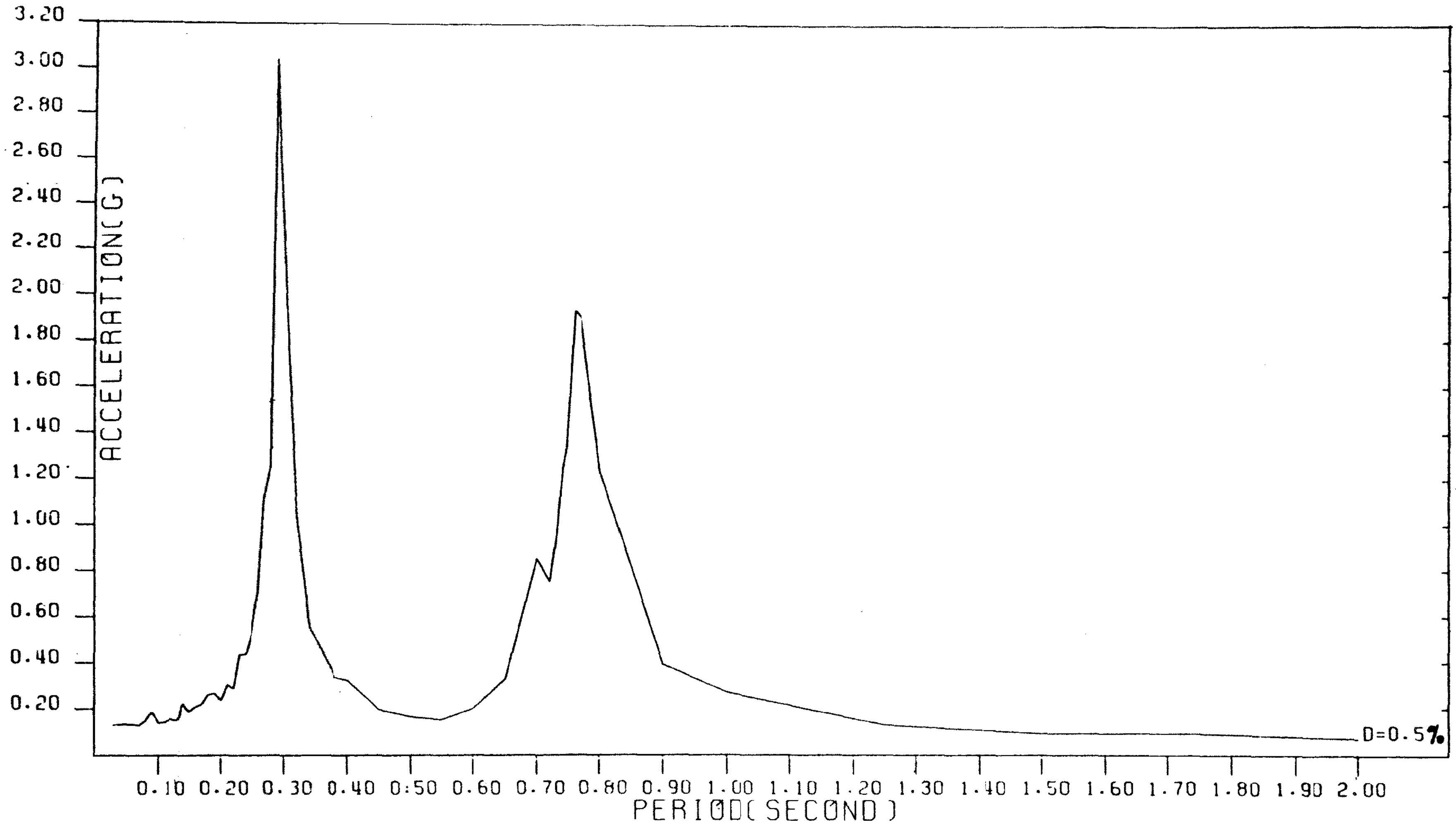
THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM
HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR BUILDING EL 24.0
HORIZONTAL FLOOR RESPONSE SPECTRA

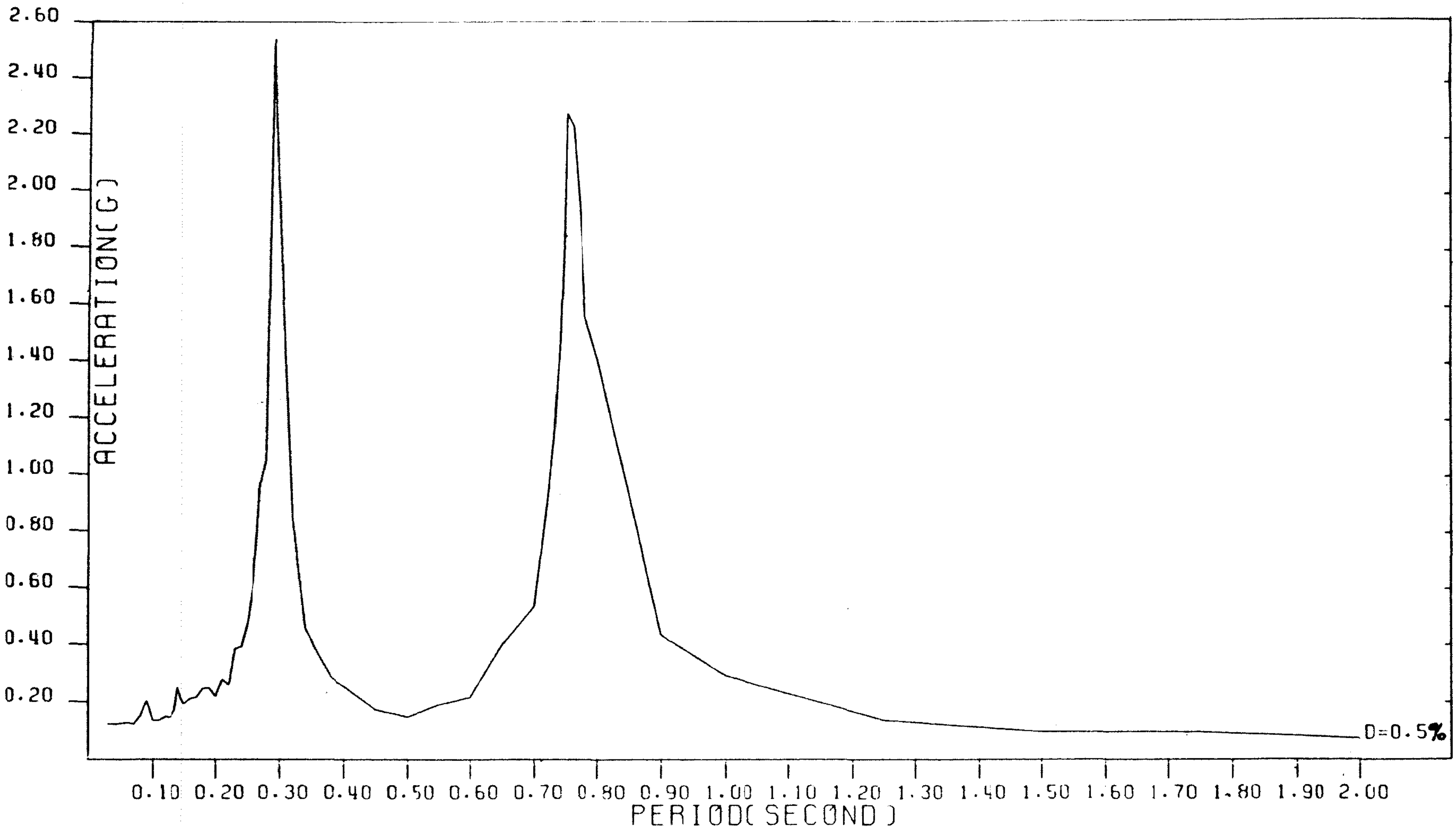
FIGURE 3.7-13



NOTE:

THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT
REACTOR BUILDING EL 44.0 HORIZONTAL FLOOR RESPONSE SPECTRA
FIGURE 3.7-14



NOTE:

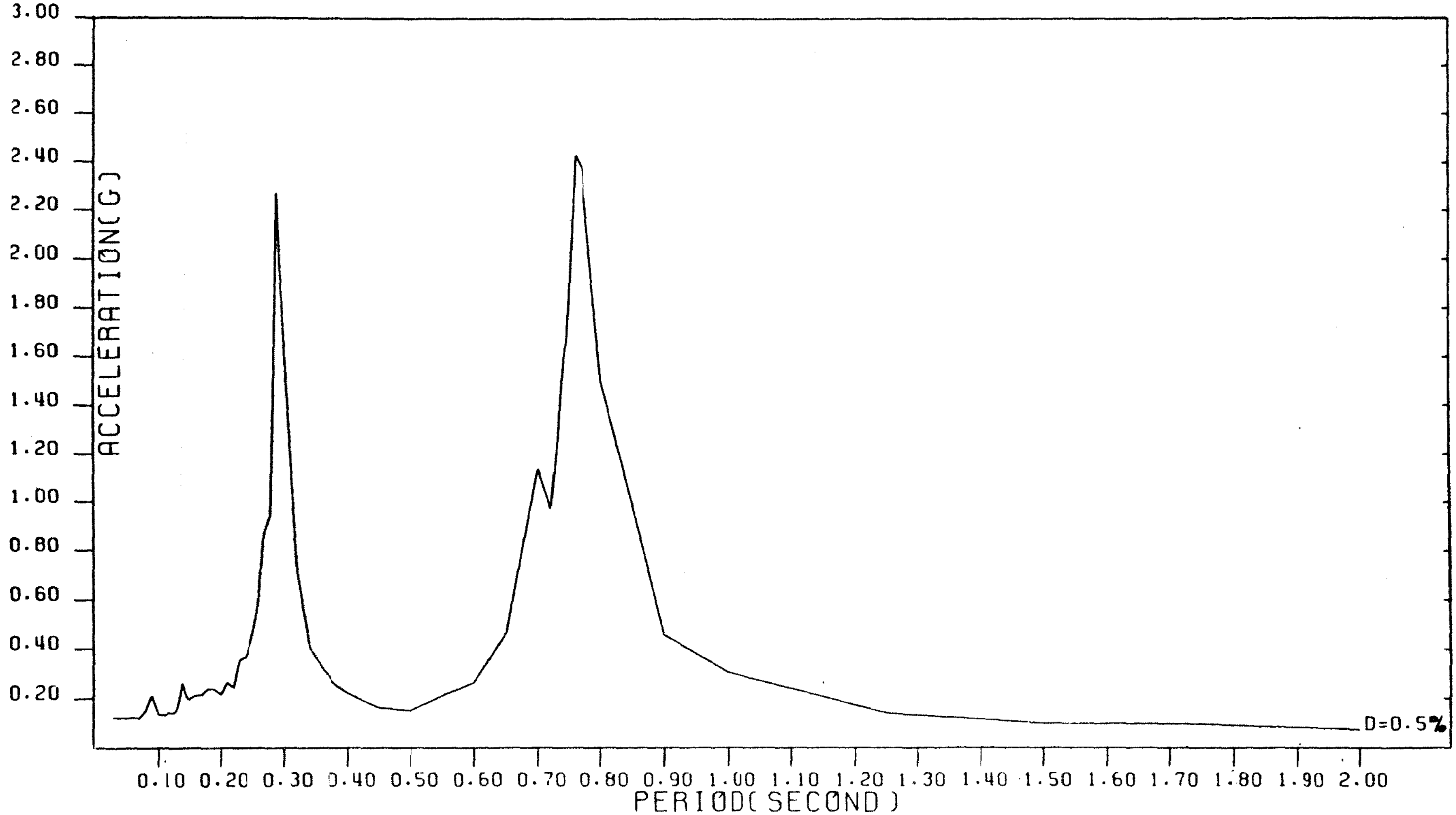
**THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM
HYPHOTHETICAL EARTHQUAKE.**

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR BUILDING EL 60.0
HORIZONTAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-15



NOTE:

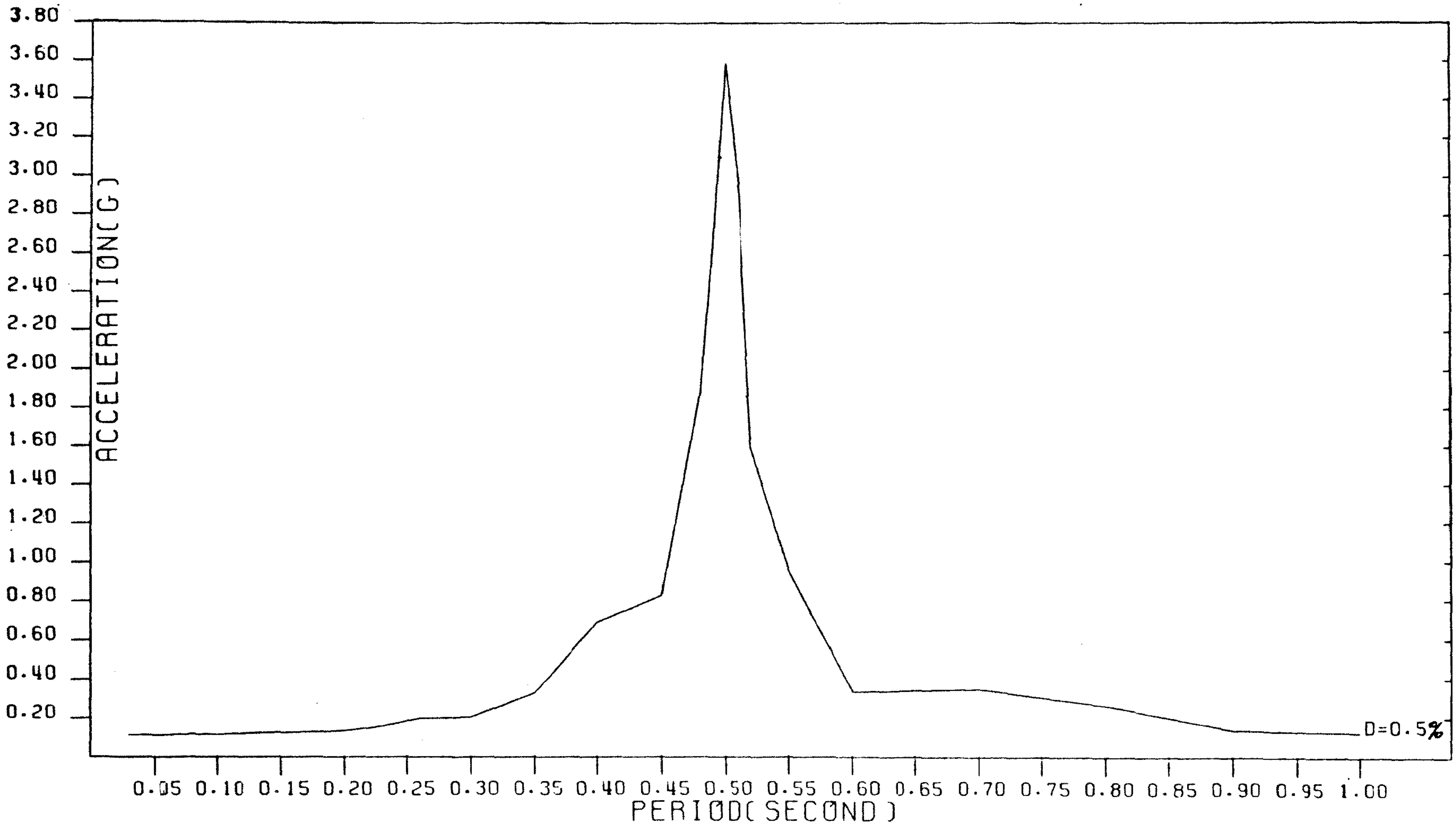
THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR BUILDING EL 68.5
HORIZONTAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-16



NOTE:

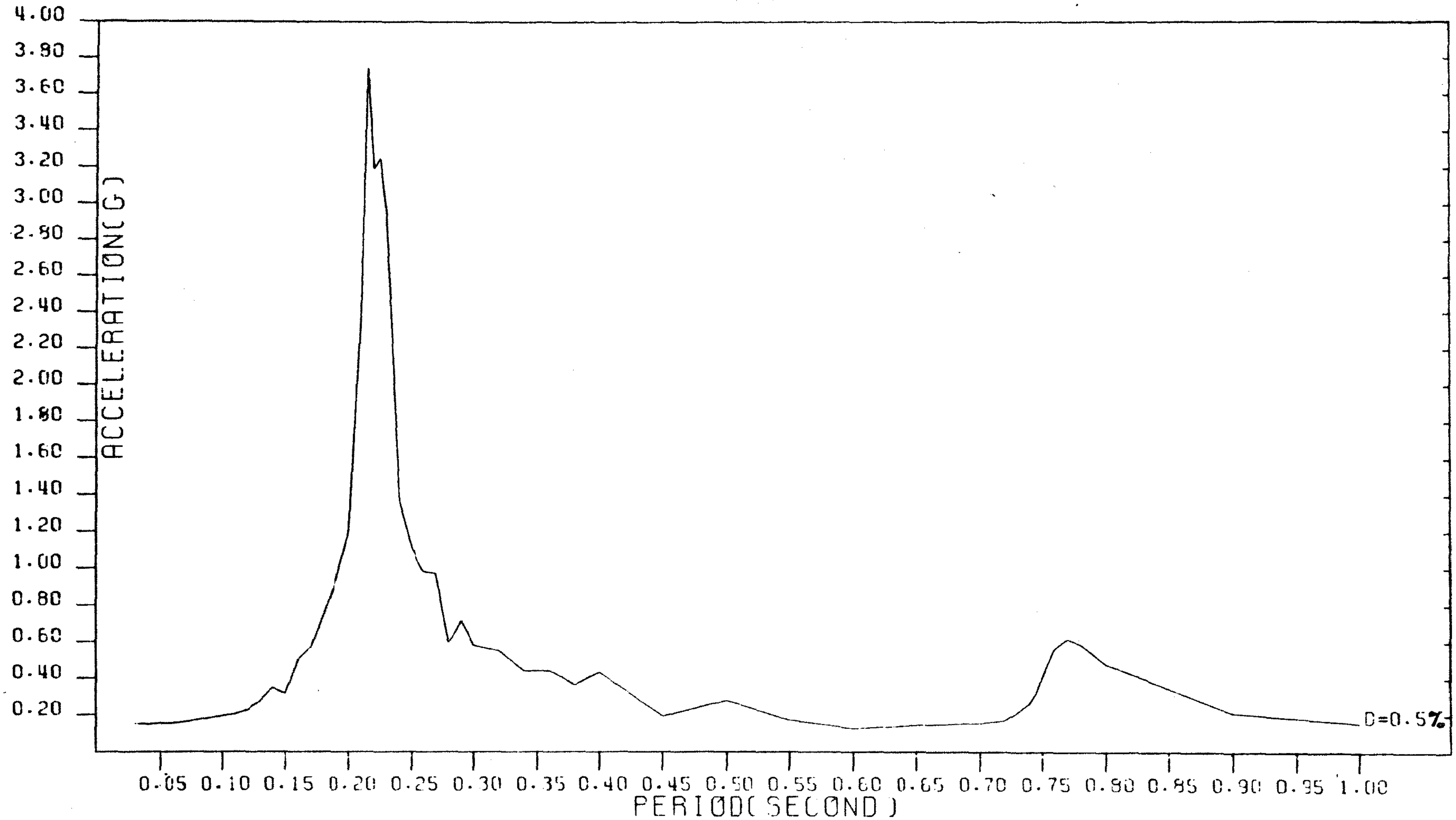
THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR BUILDING
VERTICAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-17



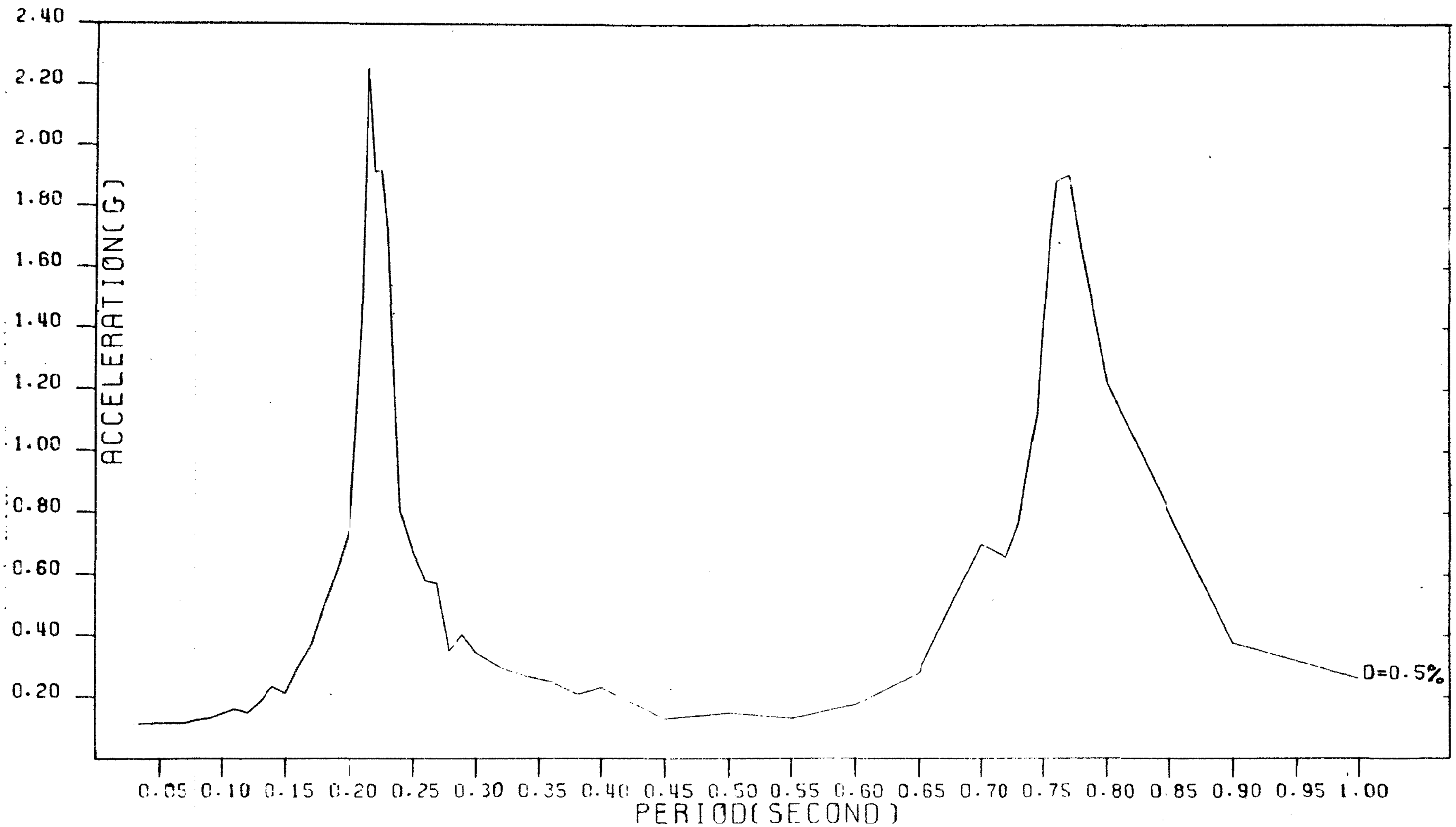
NOTE:

THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM
HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR AUX. BLDG. EL -0.5
HORIZONTAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-18



NOTE:

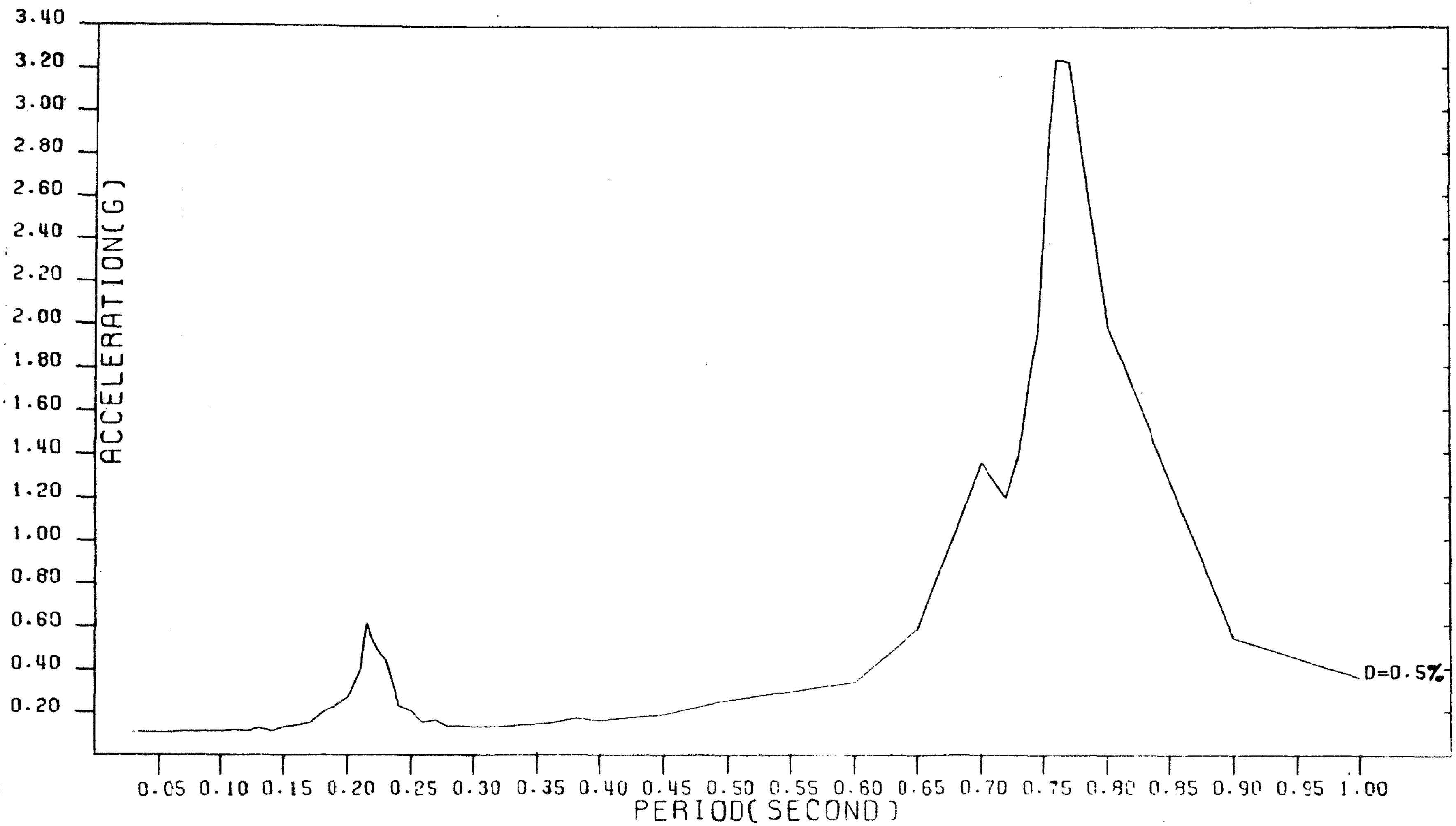
THIS SPECTRA IS FOR DESIGN EARTHQUAKE. DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR AUXILIARY BUILDING EL 19.5
HORIZONTAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-19



NOTE:

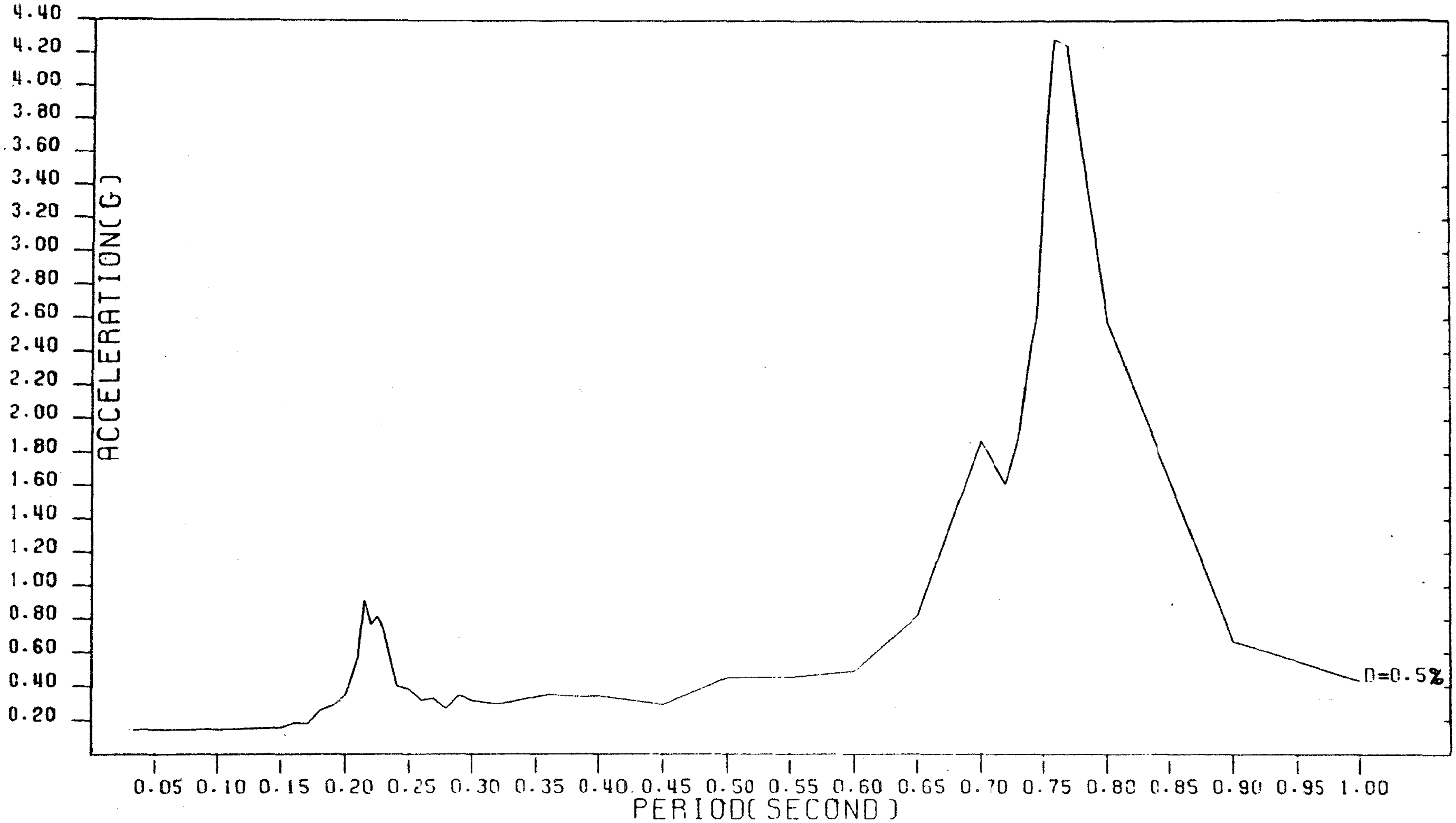
THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR AUX. BLDG. EL 43.0
HORIZONTAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-20



NOTE:

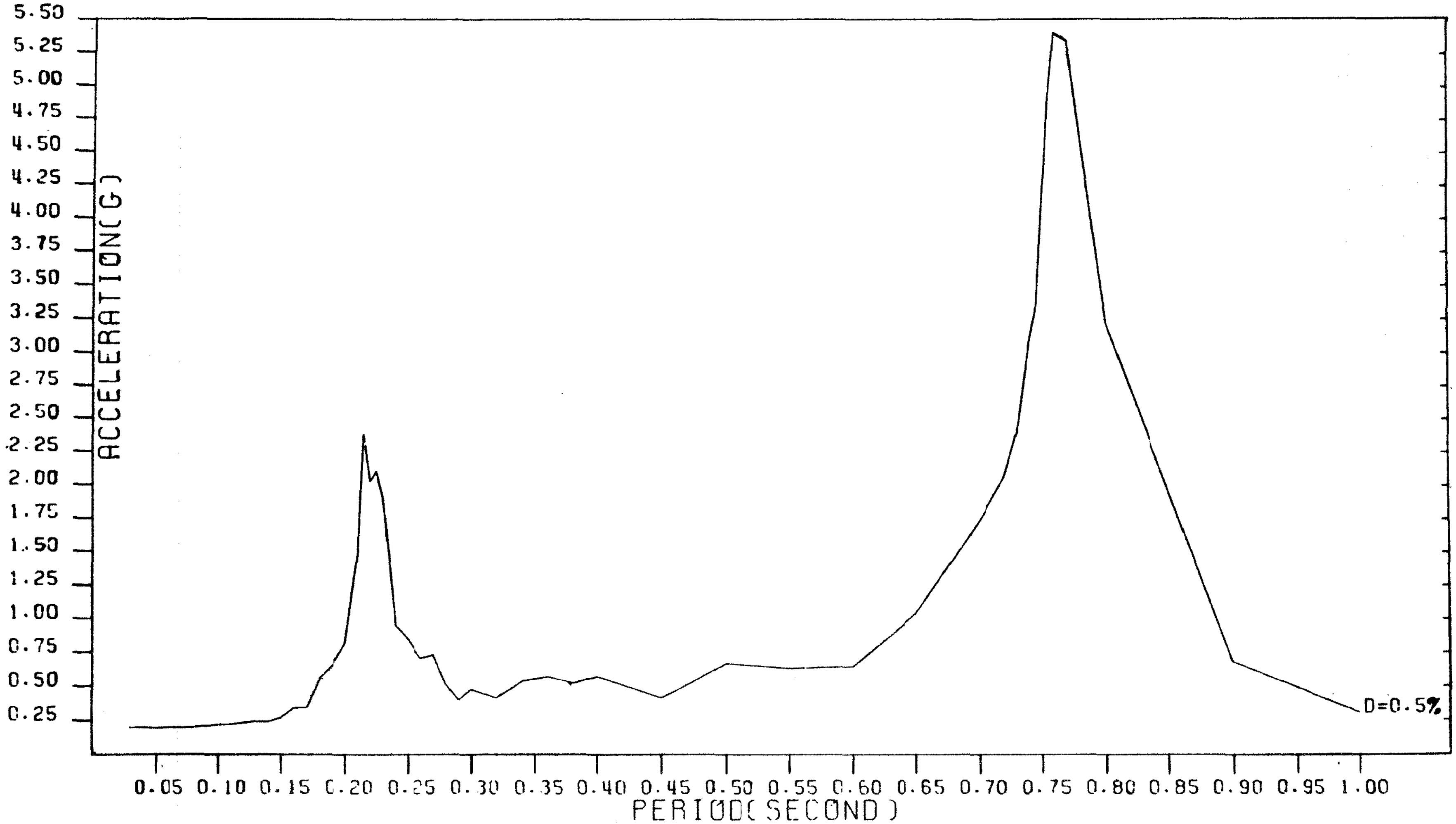
THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR AUX. BLDG. EL 62.0
HORIZONTAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-21



NOTE:

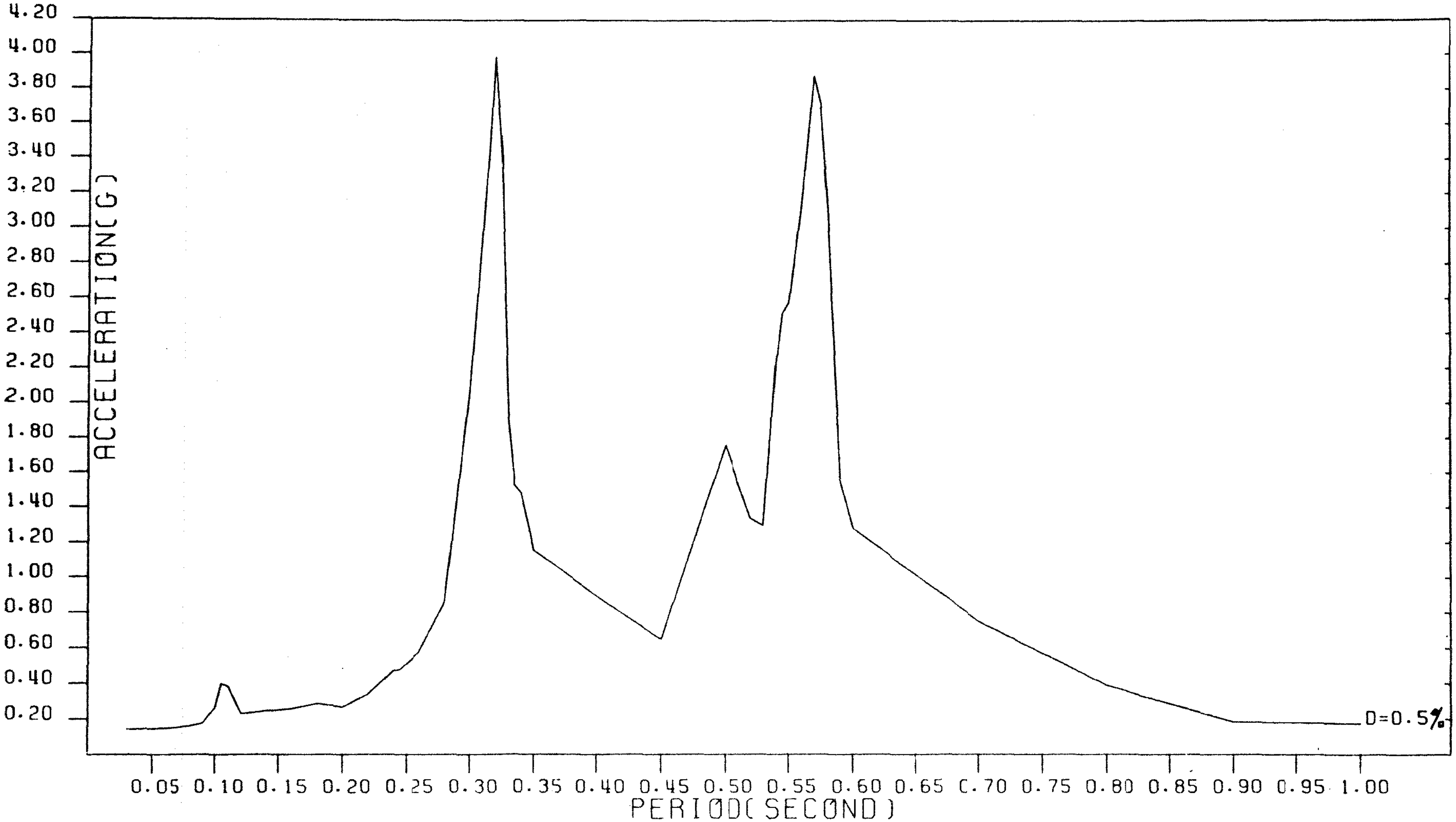
THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR AUX. BLDG. EL 82.0
HORIZONTAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-22



NOTE:

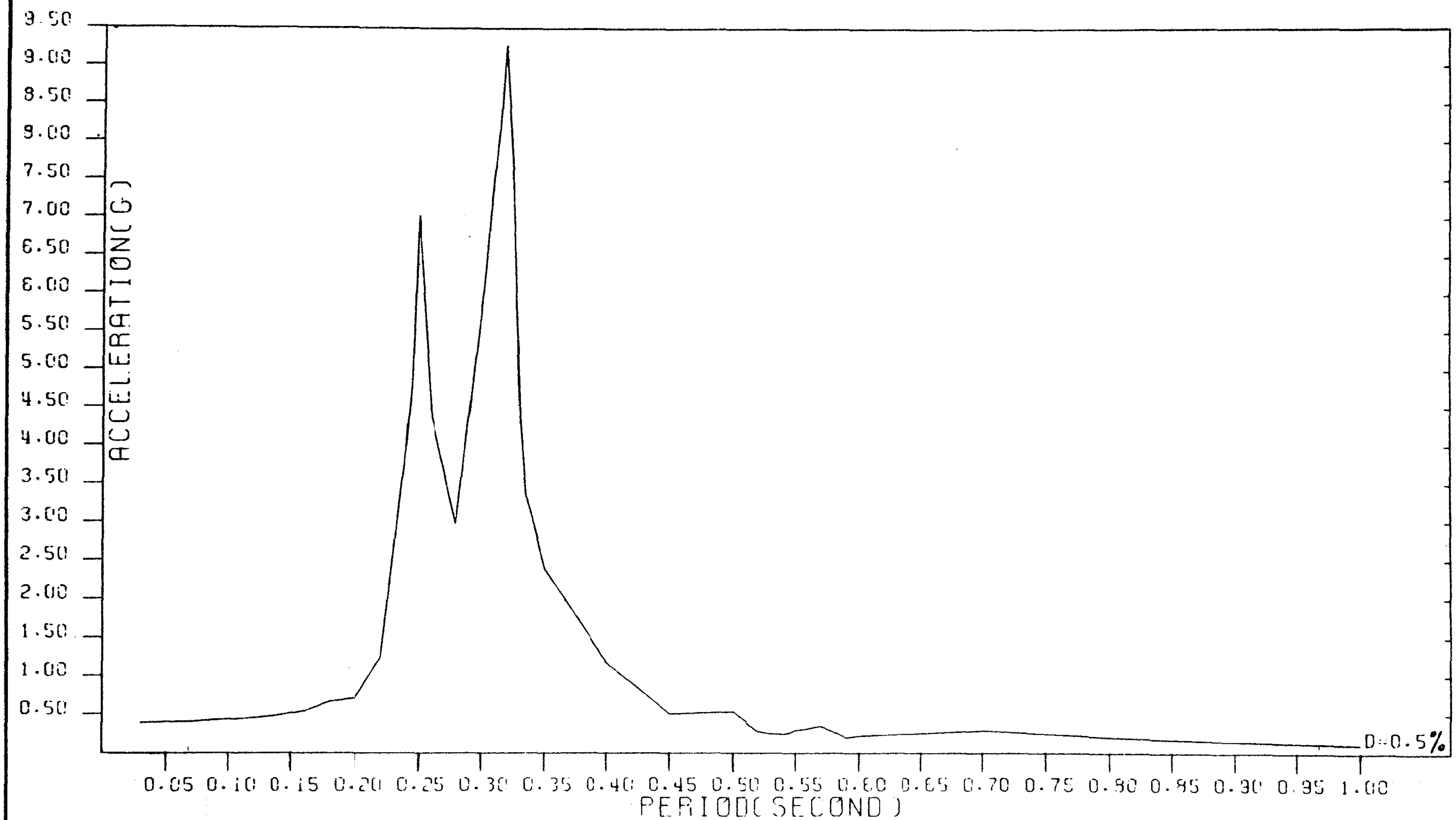
THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR AUX. BLDG.
EL 62.0, 43.0, 19.5, −0.5, −10.0
VERTICAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-23



NOTE:

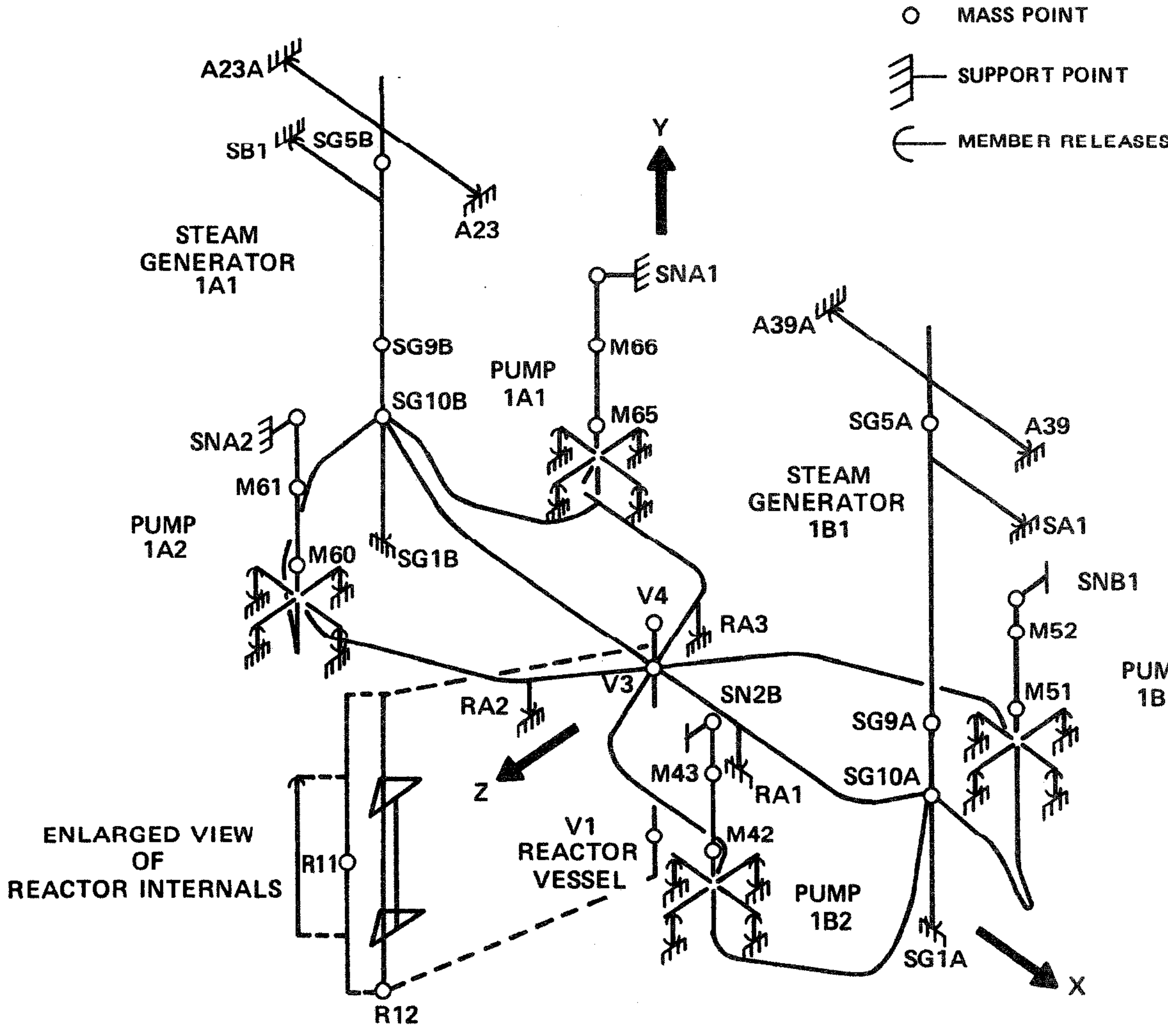
THIS SPECTRA IS FOR DESIGN EARTHQUAKE.
DOUBLE ACCELERATION VALUES FOR MAXIMUM HYPHOTHETICAL EARTHQUAKE.

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

REACTOR AUX. BLDG. EL 82.0
VERTICAL FLOOR RESPONSE SPECTRA

FIGURE 3.7-24



COMPONENT NAME	MASS POINT NAME	MASS POINT DEGREES OF FREEDOM
REACTOR	V4, RI1, RI2	X, Y, Z
	V1, V3	X, Z
STEAM GENERATORS	SG5A, SG5B	X, Y, Z
	SG9A, SG9B SG10A, SG10B	X, Z
REACTOR COOLANT PUMPS	M43, M52	X, Y, Z
	M61, M66	
	M42, M51	X, Z
	M60, M65	

COMPONENT NAME	SUPPORT NAME	SUPPORT RESTRAINT DIRECTIONS
REACTOR	RA1	Fy, Fz
	RA2, RA3	Fy, FH*
STEAM GENERATORS	SG1A, SG1B	Fy, Fz, My
	SA1, SB1	Fx
	A23, A23A	Fz
	A39, A39A	
REACTOR COOLANT PUMPS	(VERTICAL SUPPORTS 4 PLACES ON EACH PUMP)	Fy
	SNA1, A2, B1, B2	Fx, Fz

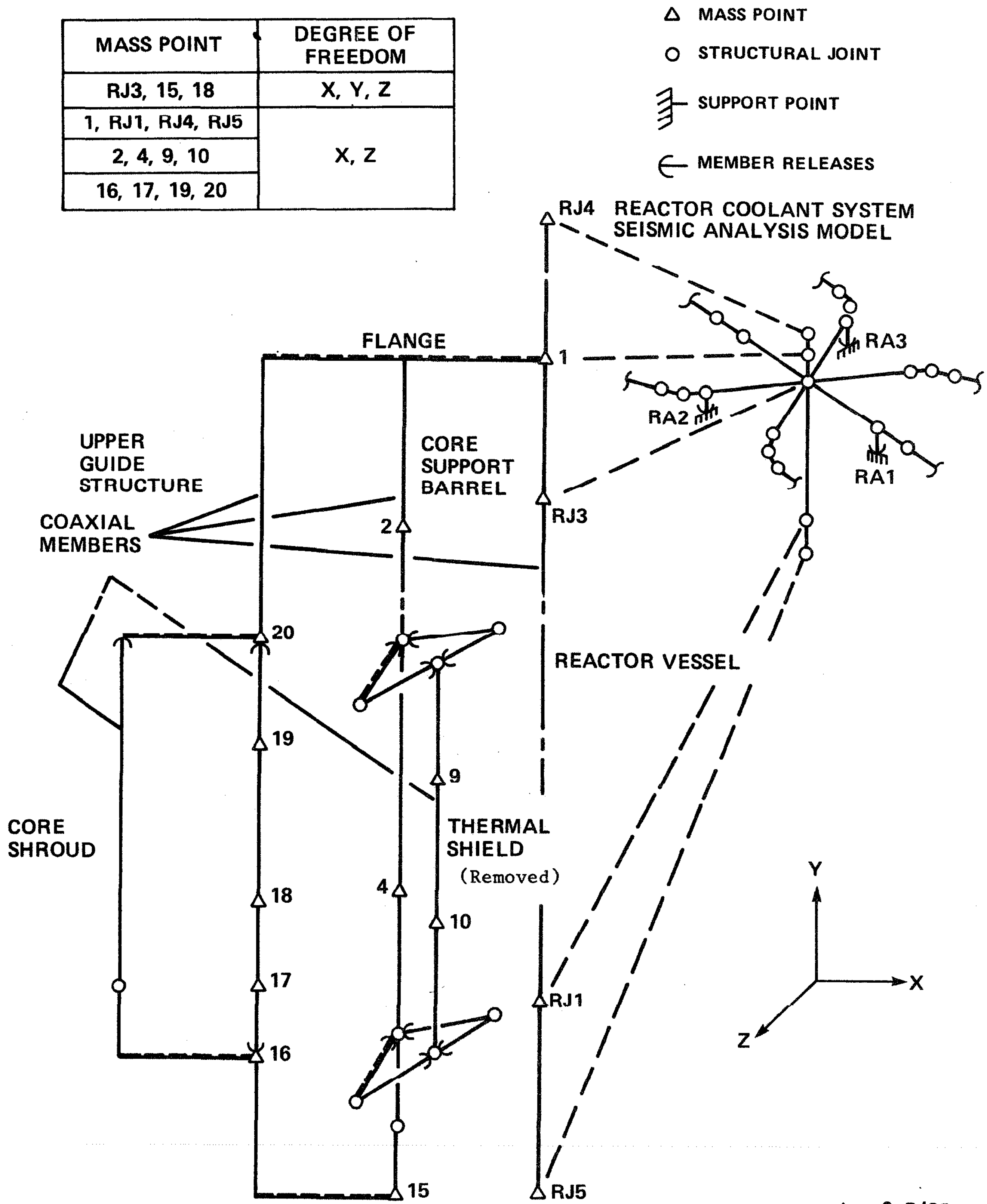
*FH RESTRAINTS ARE PERPENDICULAR TO THE REACTOR VESSEL RADIAL DIRECTION IN THE HORIZONTAL PLANE.

Am. 3-7/85

FOR HISTORICAL INFORMATION ONLY

FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	TYPICAL REACTOR COOLANT SYSTEM SEISMIC ANALYSIS MODEL	Figure 3.7-25

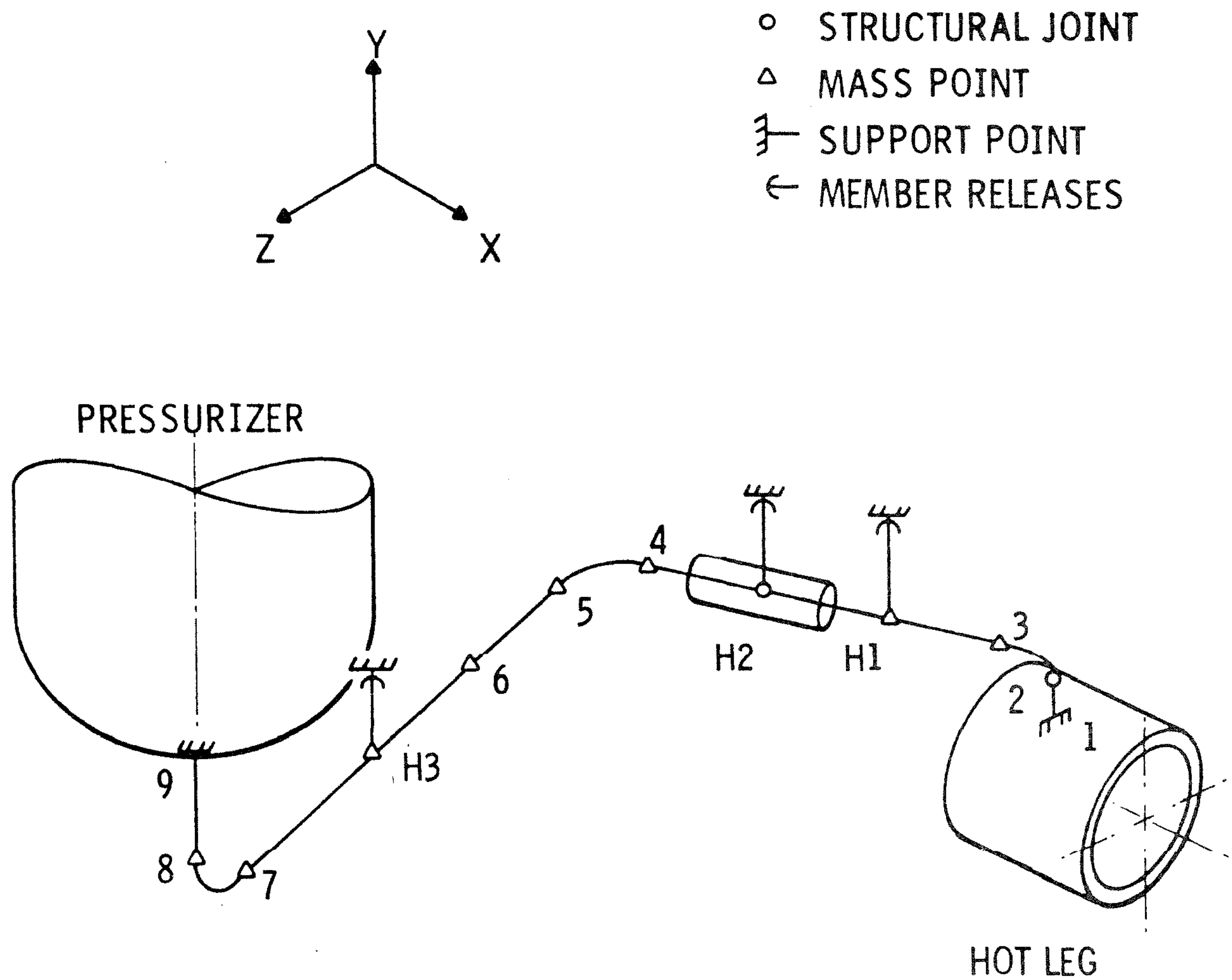
MASS POINT	DEGREE OF FREEDOM
RJ3, 15, 18	X, Y, Z
1, RJ1, RJ4, RJ5	X, Z
2, 4, 9, 10	
16, 17, 19, 20	



Am. 3-7/85

FOR HISTORICAL INFORMATION ONLY

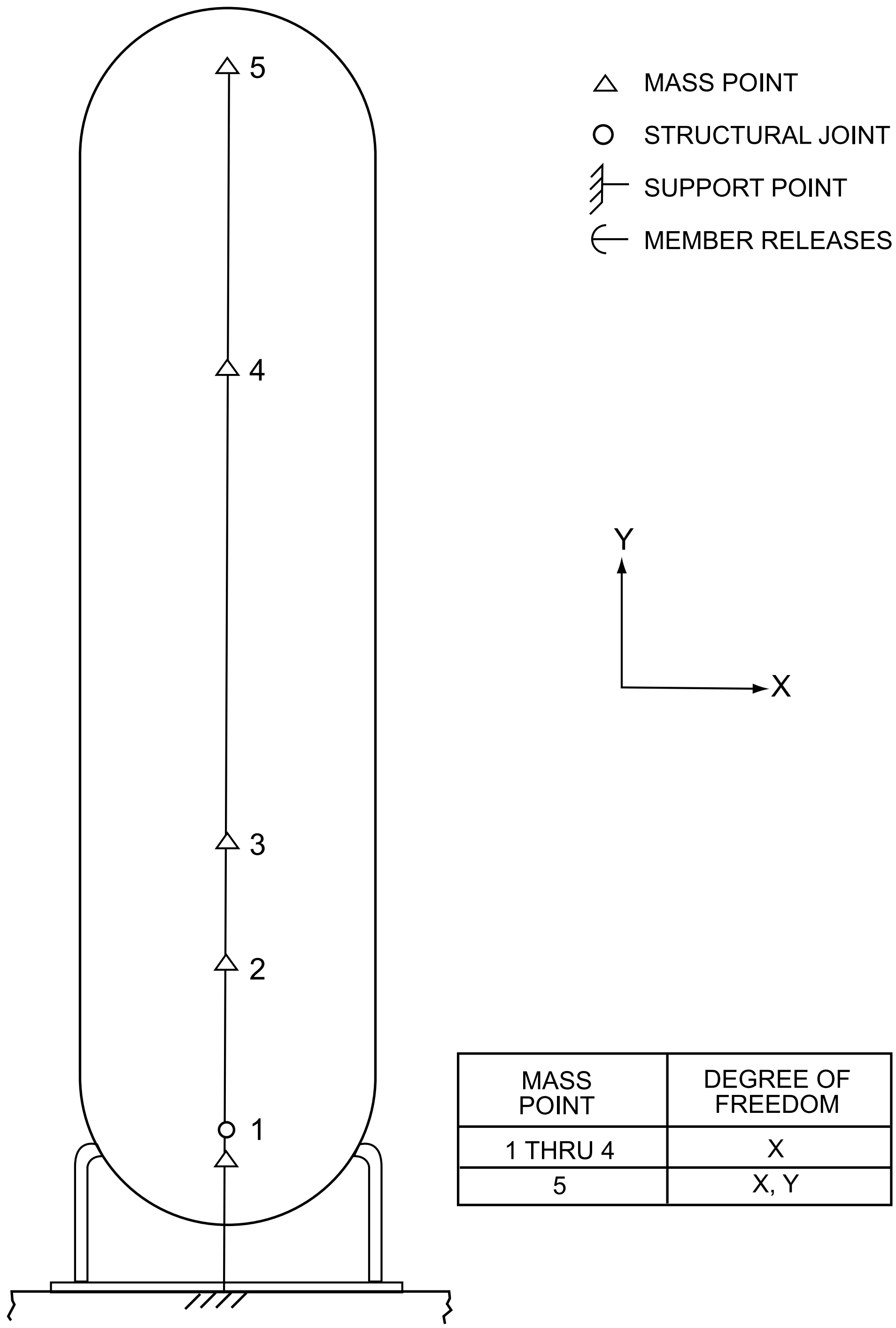
FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	(RV14) REACTOR AND INTERNALS SEISMIC ANALYSIS MODEL	Figure 3.7-26



MASS POINT	DEGREES OF FREEDOM
2-8, H1, H3	X, Y, Z

SUPPORT NAME	RESTRAINT DIRECTION
H1, H2, H3	Y

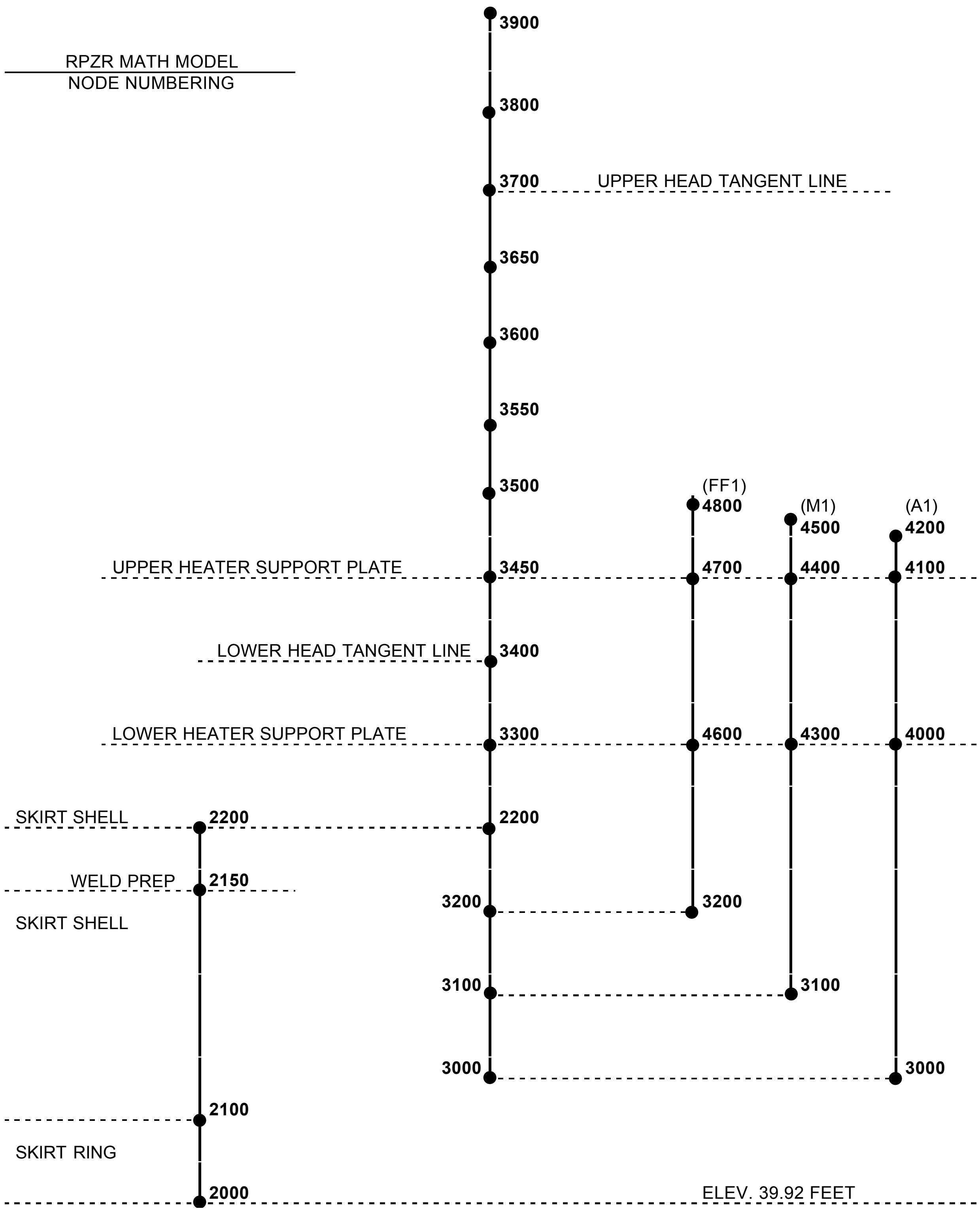
FLORIDA POWER & LIGHT CO. St. Lucie Plant	Surge Line Seismic Analysis Model	Figure 3.7-27



MASS POINT	DEGREE OF FREEDOM
1 THRU 4	X
5	X, Y

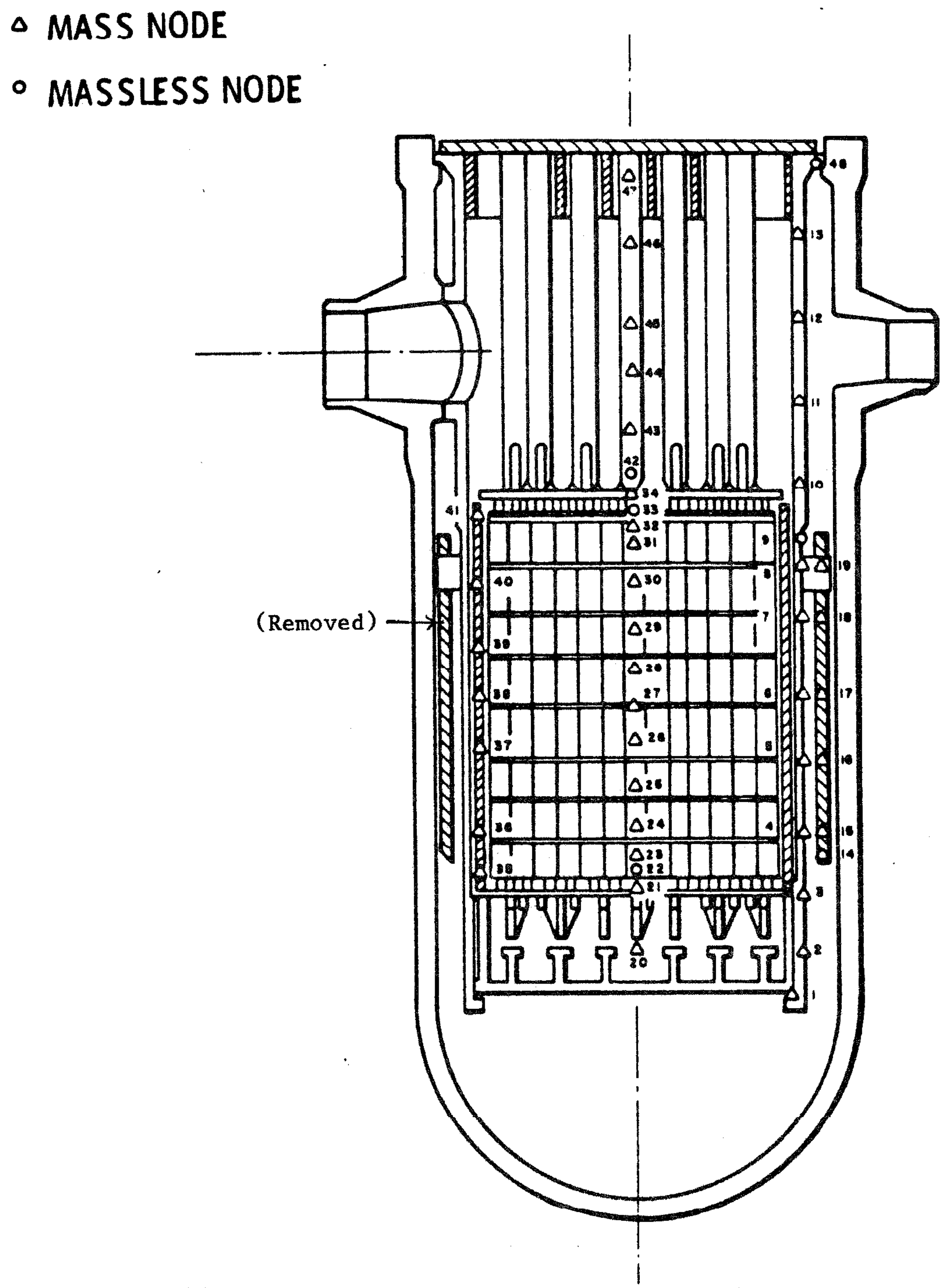
Florida Power & Light Company
St. Lucie Plant Unit 1

Original Pressurizer
Seismic Analysis Model
Figure 3.7-28a



Florida Power & Light Company
St. Lucie Plant Unit 1

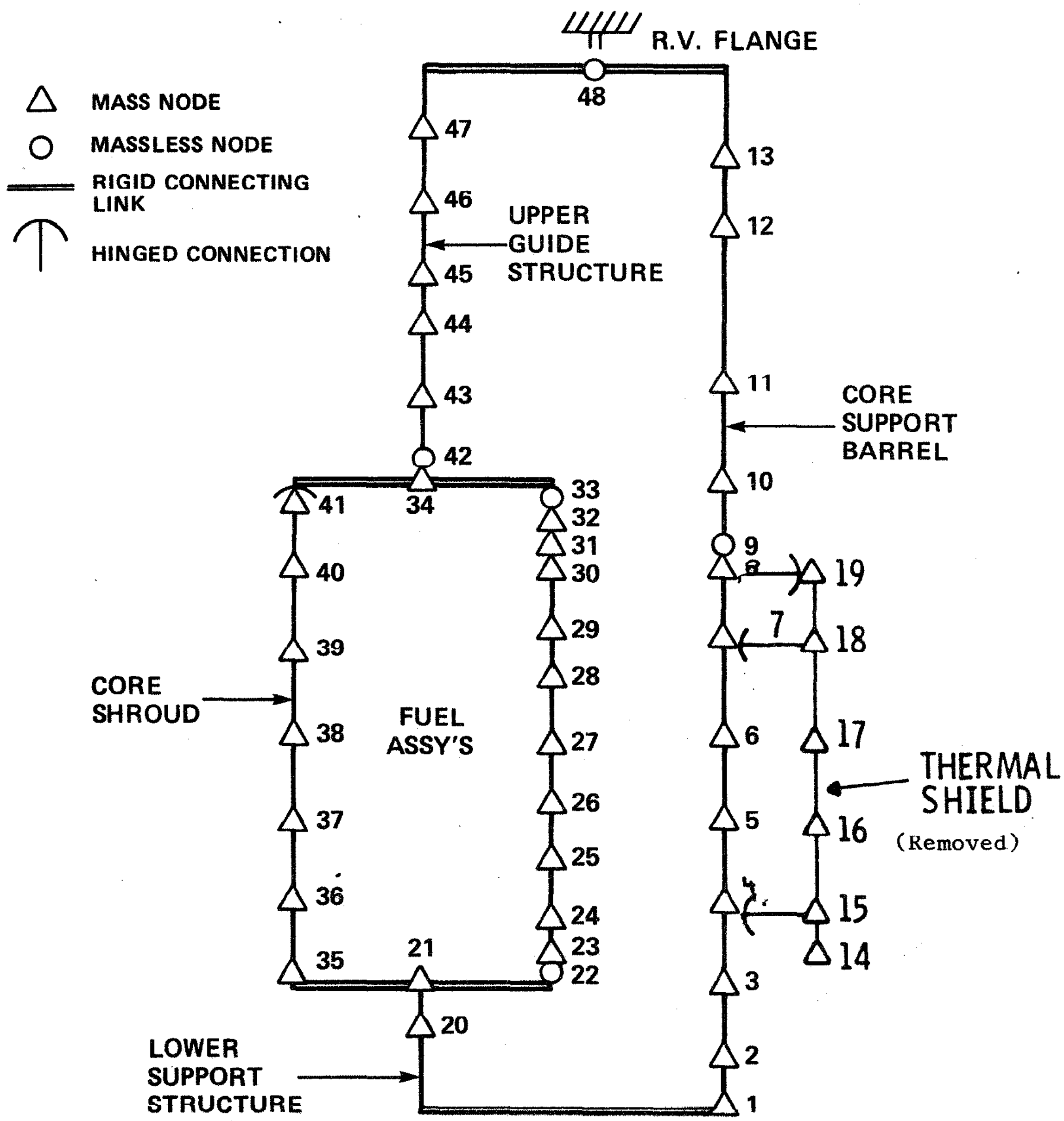
Replacement Pressurizer
Seismic Analysis Model
Figure 3.7-28b



FOR HISTORICAL INFORMATION

Am. 3-7/85

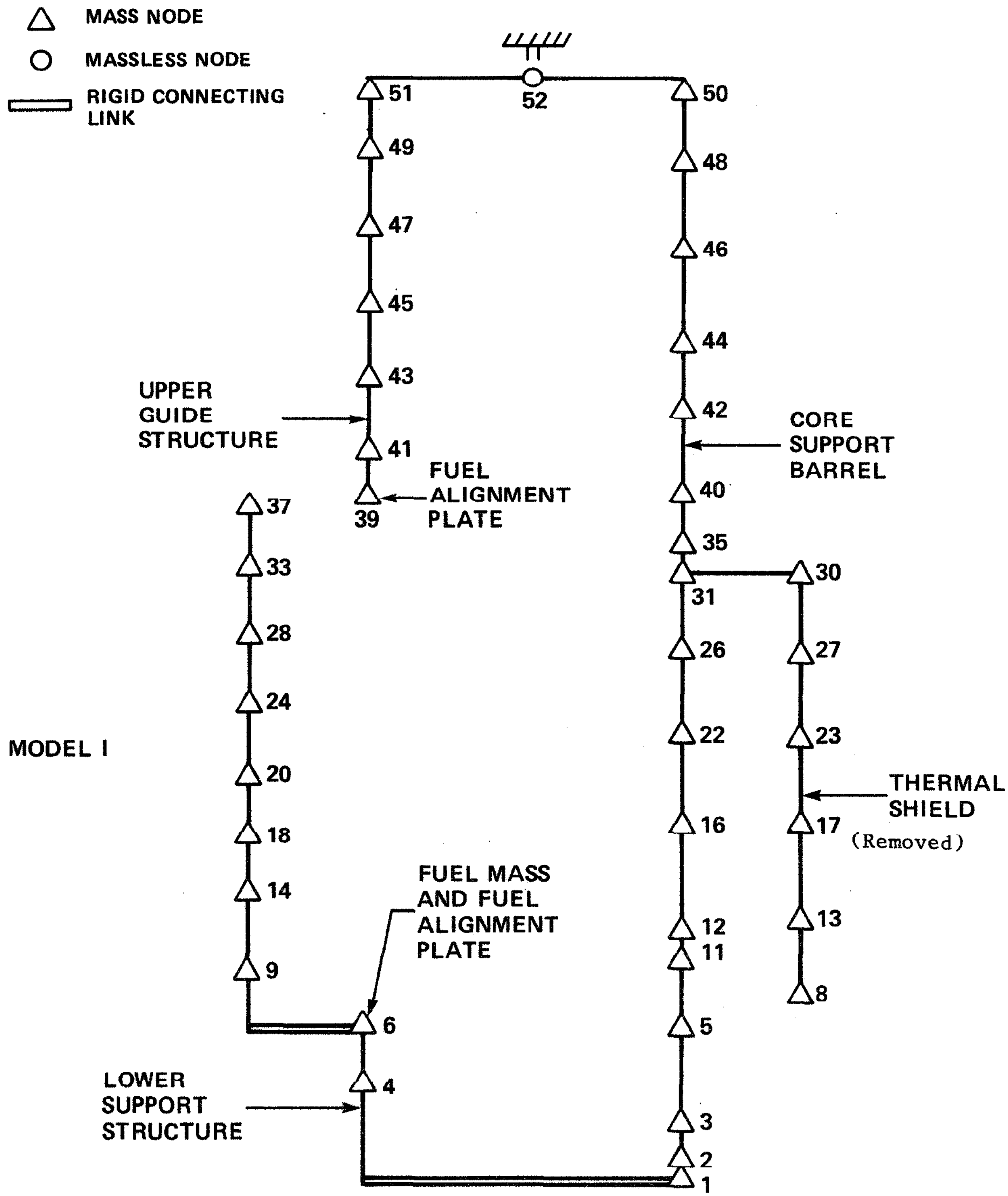
FLORIDA POWER & LIGHT CO. St. Lucie Plant	Representative Node Locations Horizontal Mathematical Model	Figure 3.7-29



FOR HISTORICAL INFORMATION

Am. 3-7/85

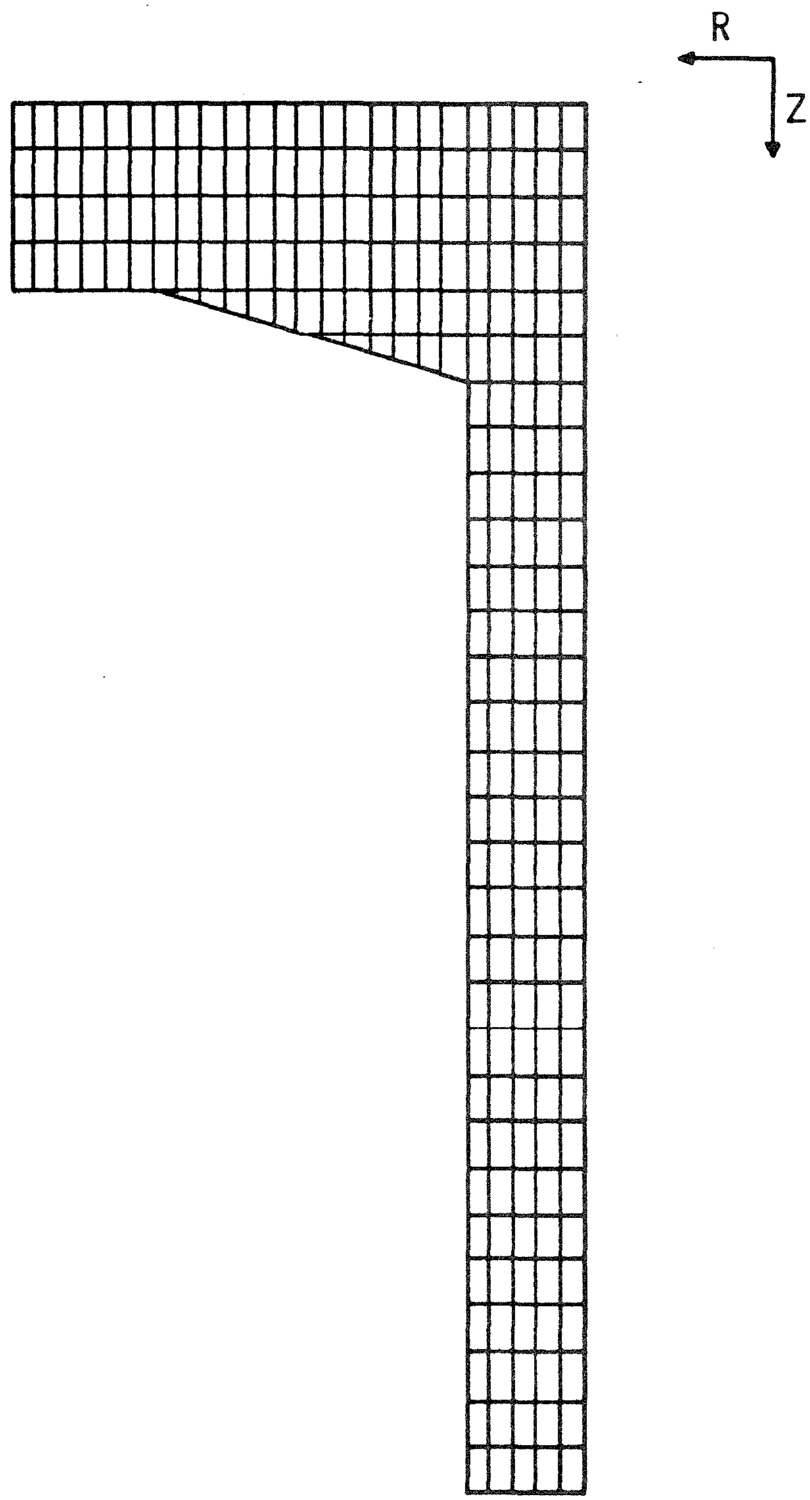
FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	MATHEMATICAL MODEL HORIZONTAL SEISMIC ANALYSIS	Figure 3.7-30



Am. 3-7/85

FOR HISTORICAL INFORMATION ONLY

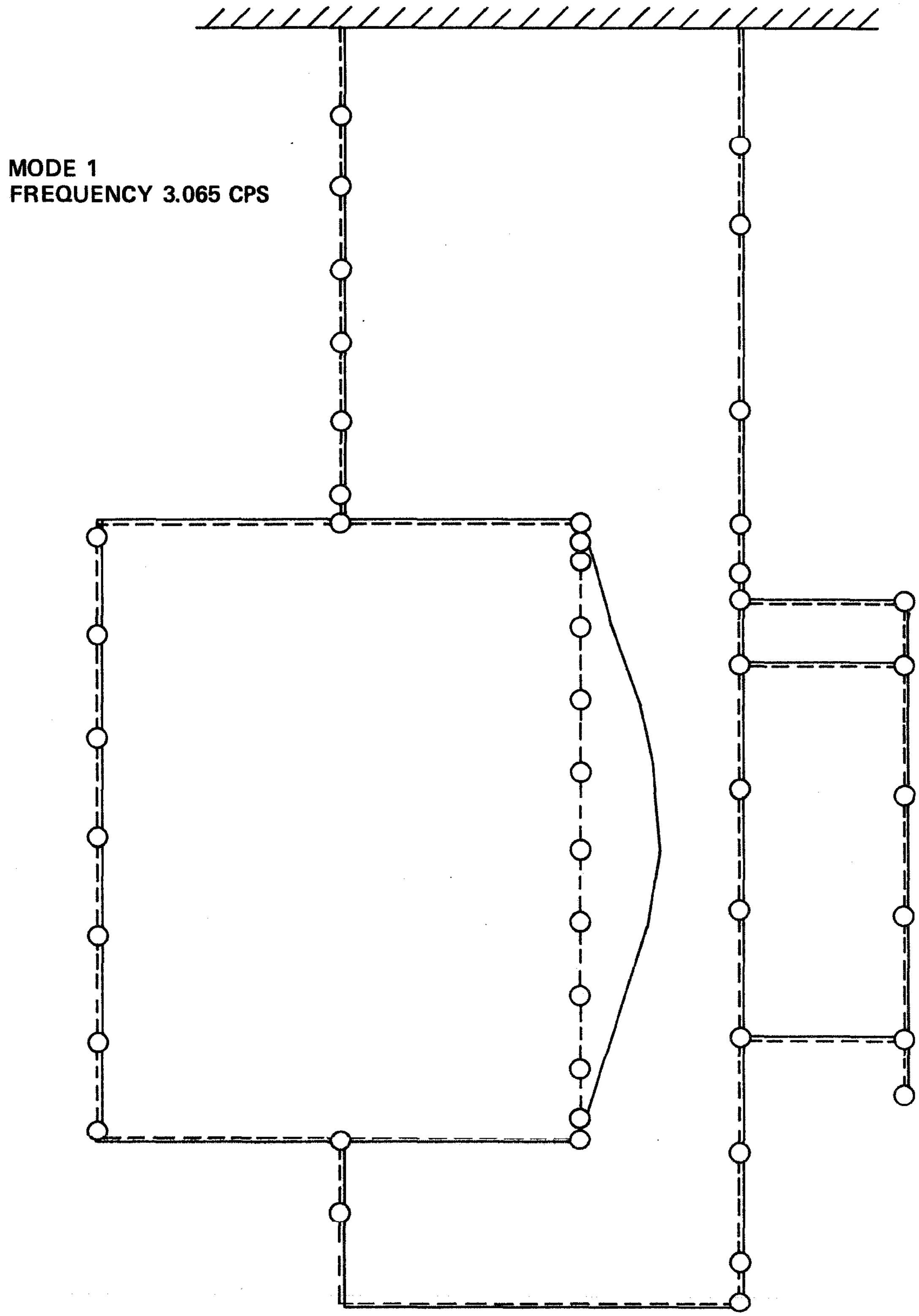
FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	MATHEMATICAL MODEL VERTICAL SEISMIC ANALYSIS	Figure 3.7-31



FLORIDA
POWER & LIGHT CO.
St. Lucie Plant

Core Support Barrel
Upper Flange
Finite Element Model

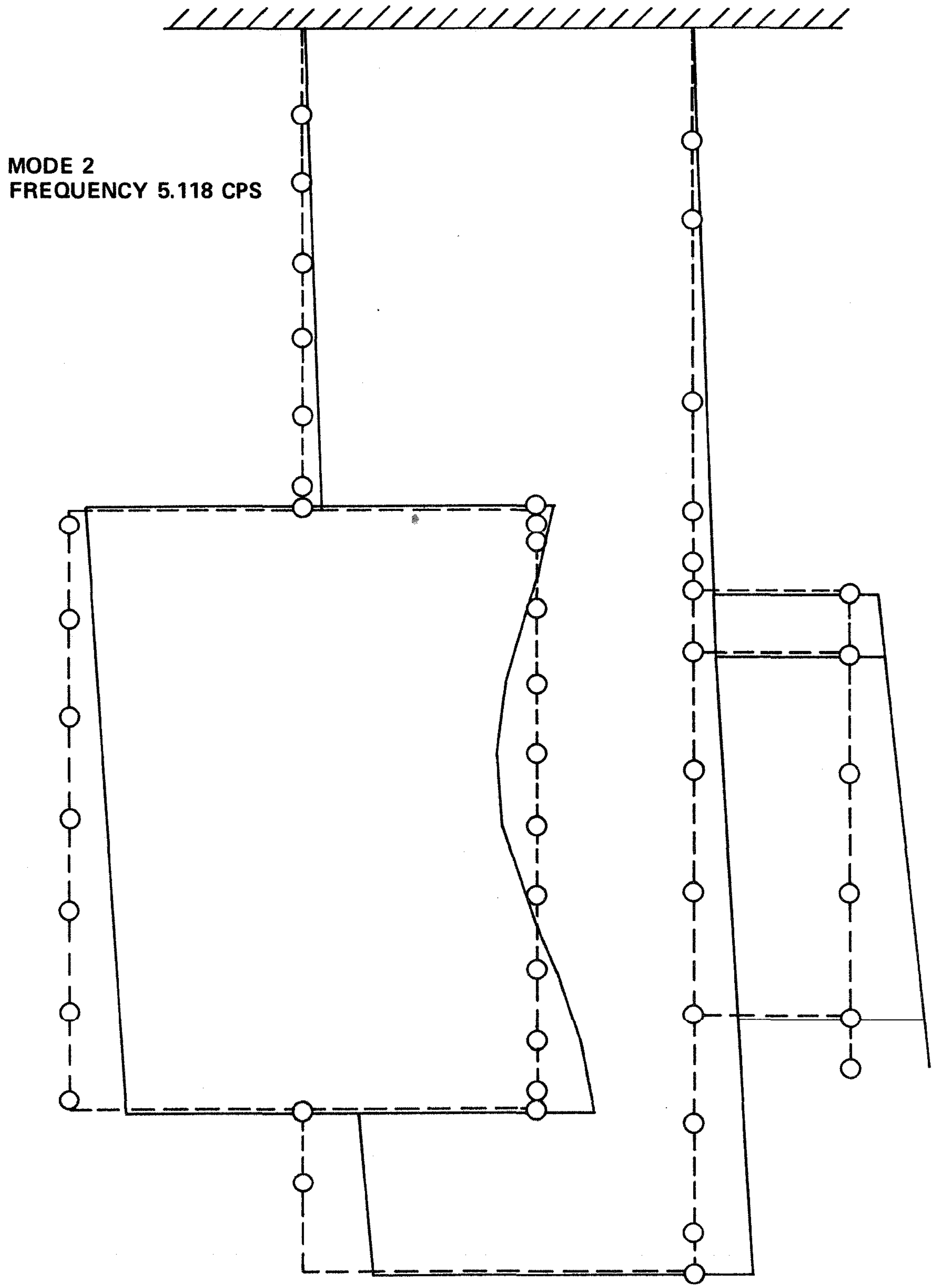
Figure
3.7-32



Am. 3-7/85

FOR HISTORICAL INFORMATION ONLY

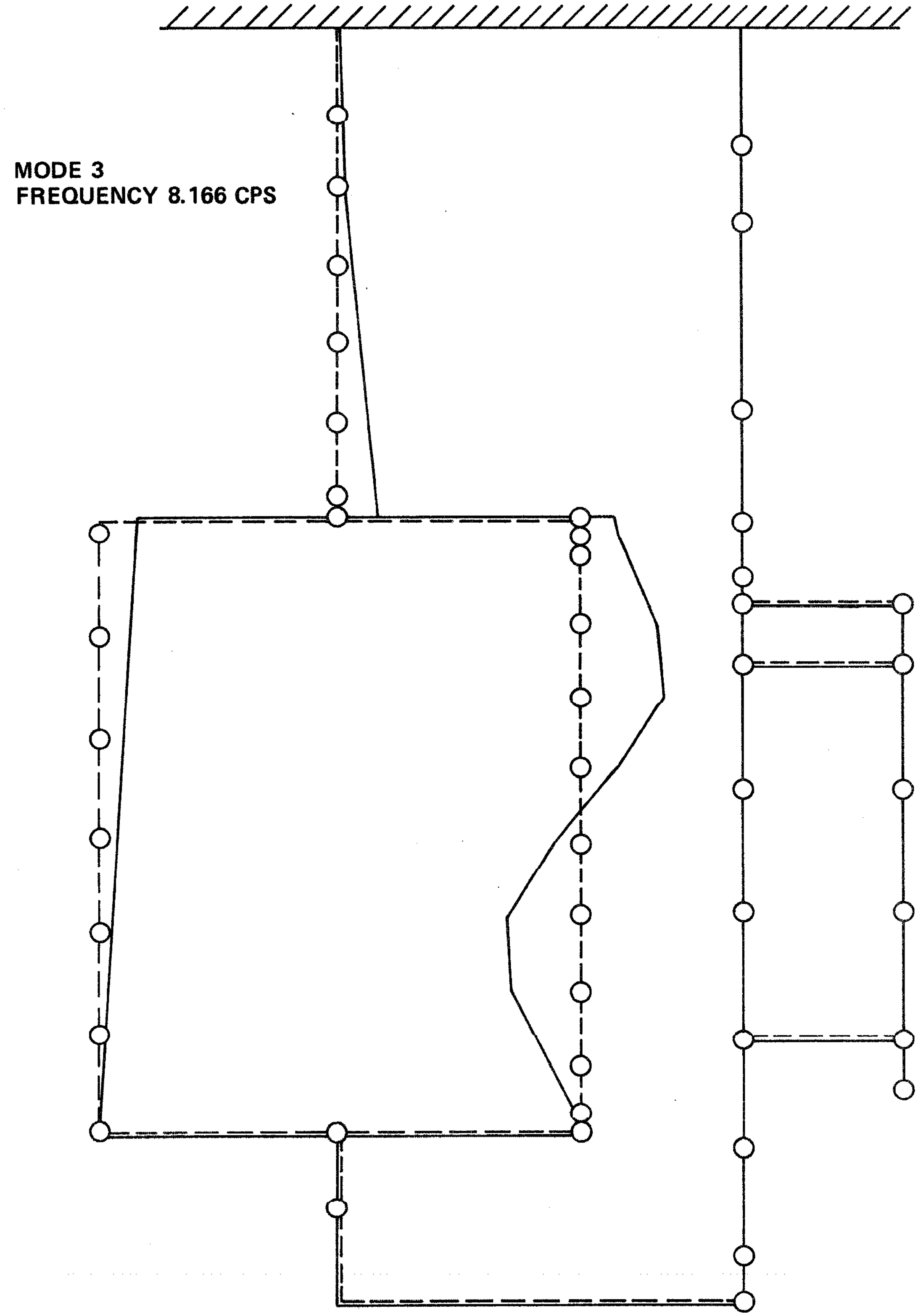
FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LATERAL SEISMIC MODEL – MODE 1, 3.065 CPS	Figure 3.7-33



Am. 3-7/85

FOR HISTORICAL INFORMATION ONLY

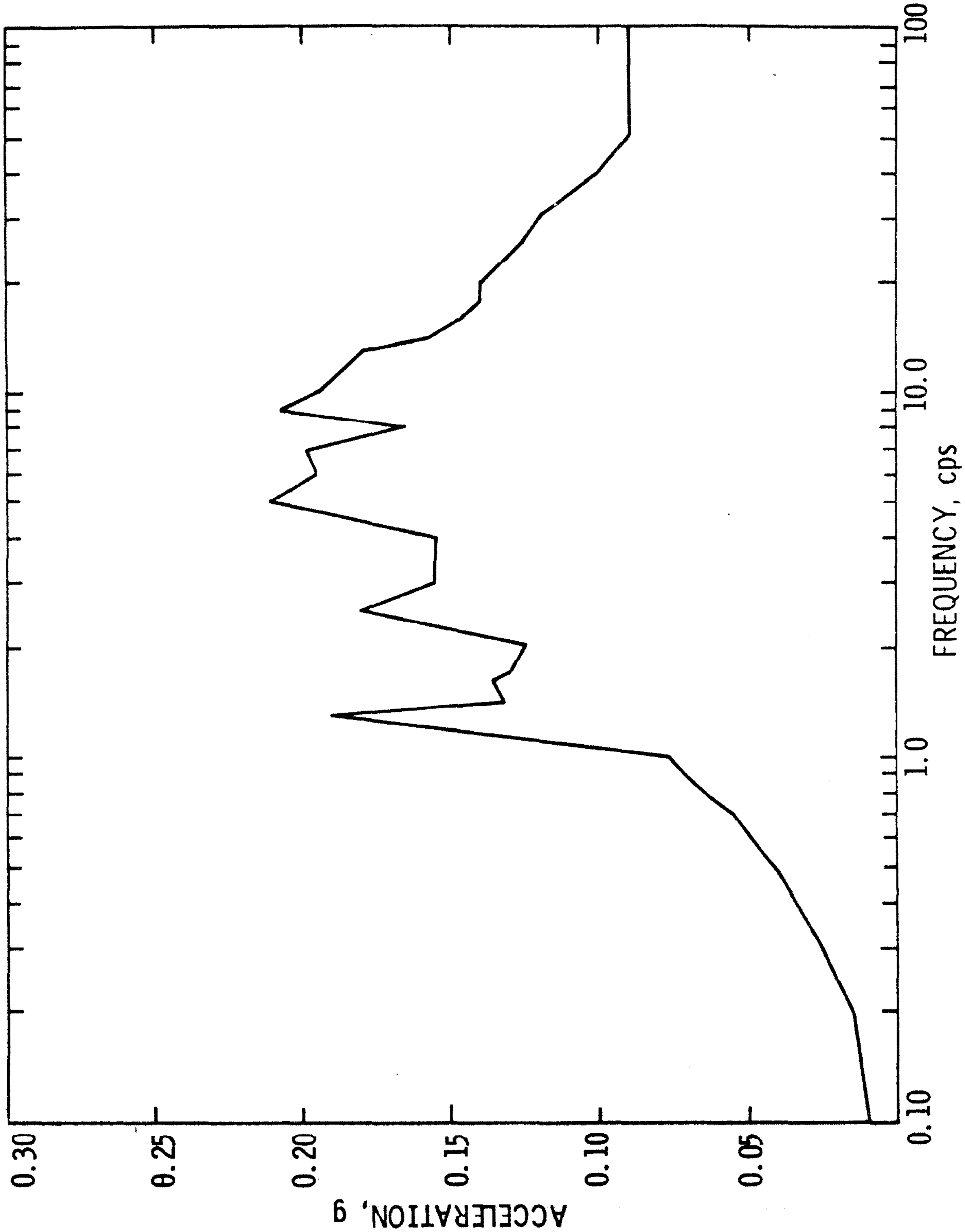
FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LATERAL SEISMIC MODEL – MODE 2, 5.118 CPS	Figure 3.7-34



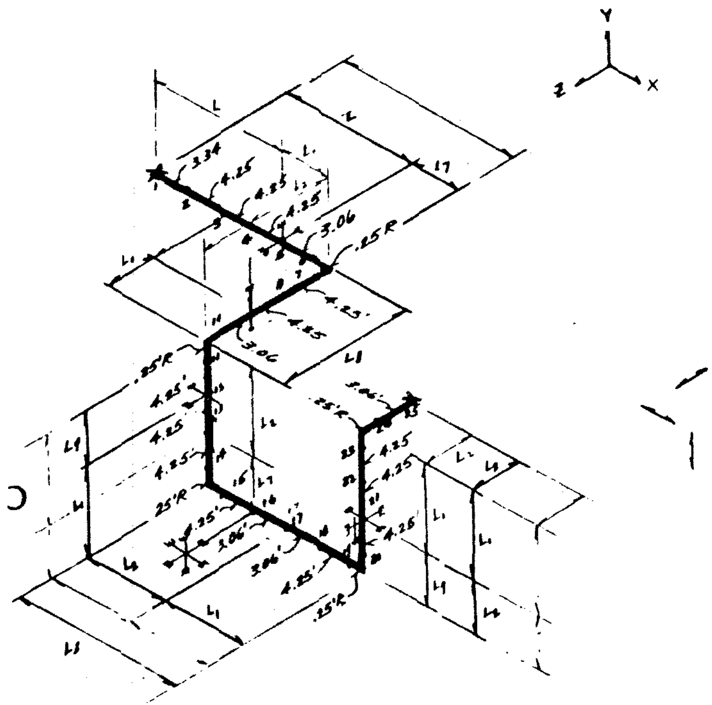
Am. 3-7/85

FOR HISTORICAL INFORMATION ONLY

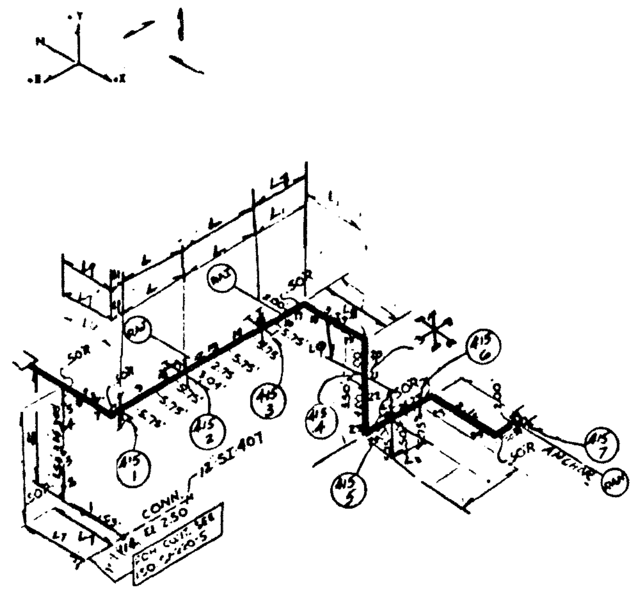
FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LATERAL SEISMIC MODEL – MODE 3, 8.166 CPS	Figure 3.7-35



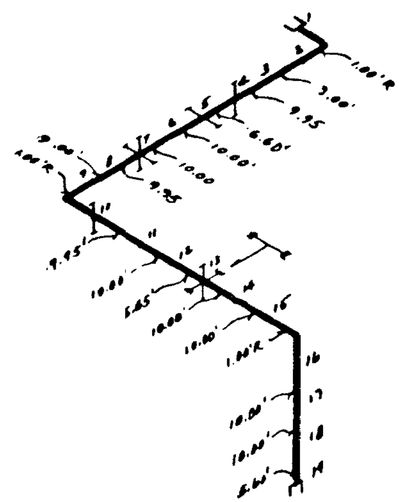
FLORIDA POWER & LIGHT CO. St. Lucie Plant	Reactor Vessel Flange Vertical Response Spectrum for the Operating Basis Earthquake (1% Damping)	Figure 3.7-36



SYSTEM 1



SYSTEM 2

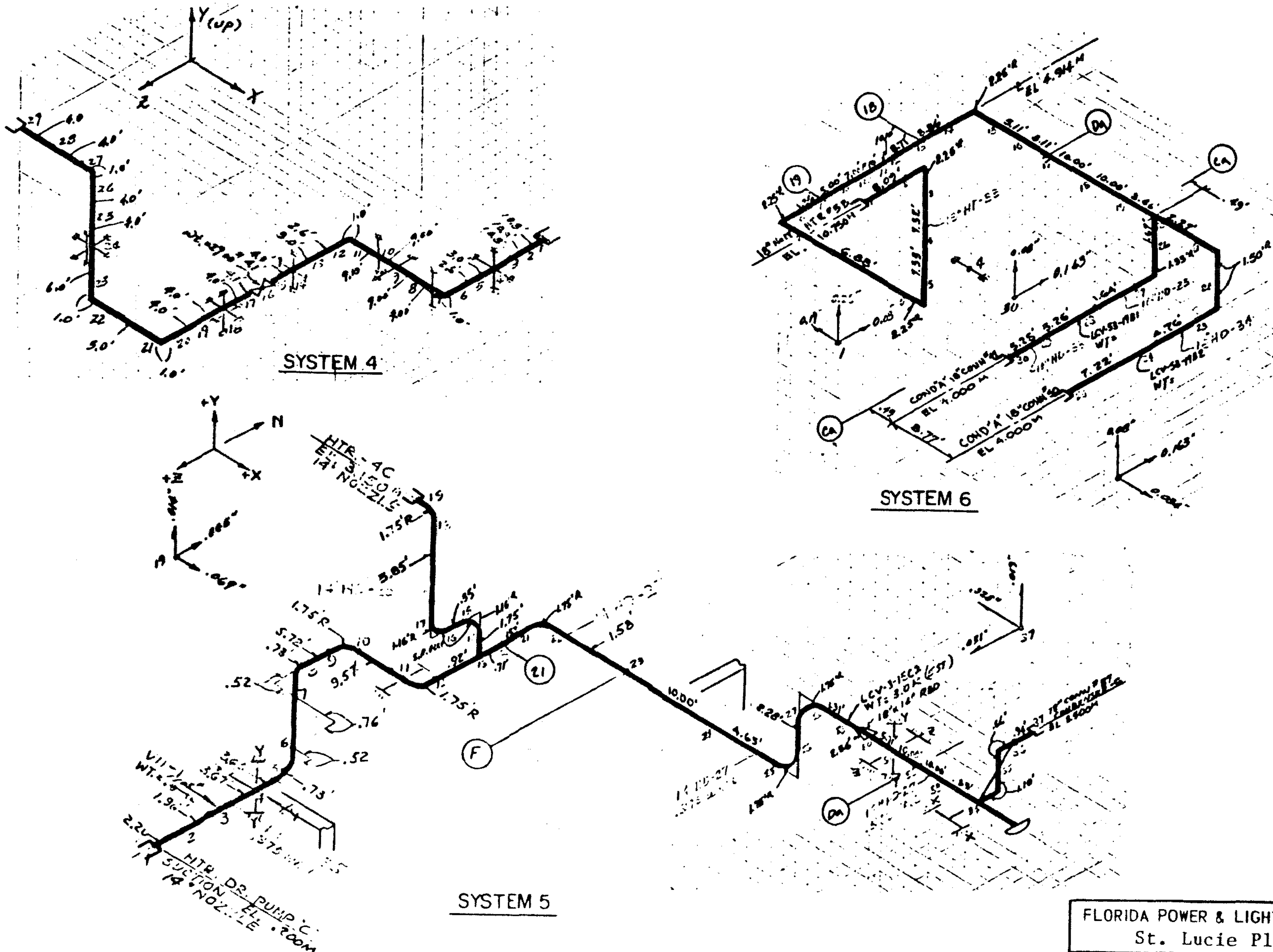


SYSTEM 3

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

SAMPLE PIPING SYSTEMS

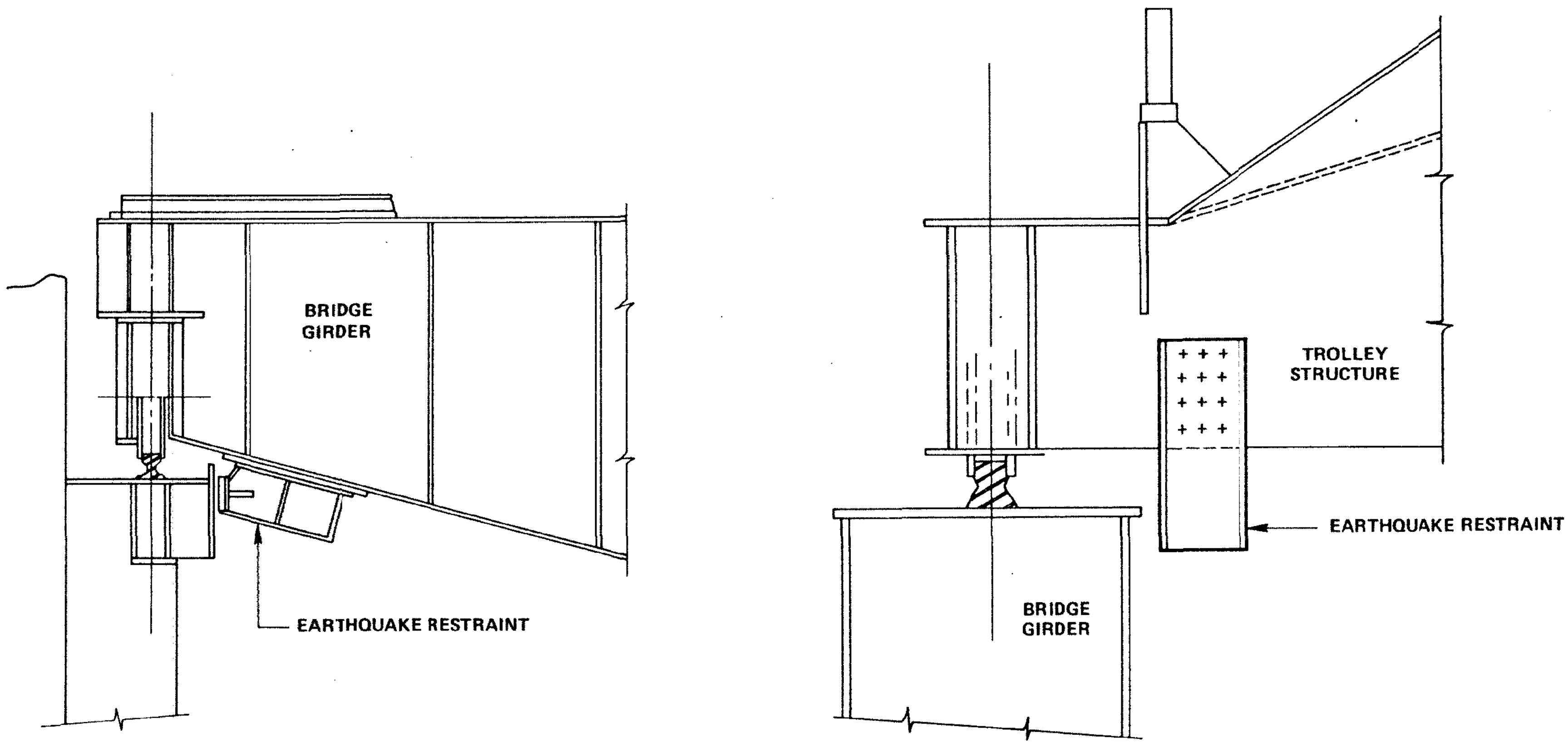
FIGURE 3.7-37



FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

SAMPLE PIPING SYSTEMS

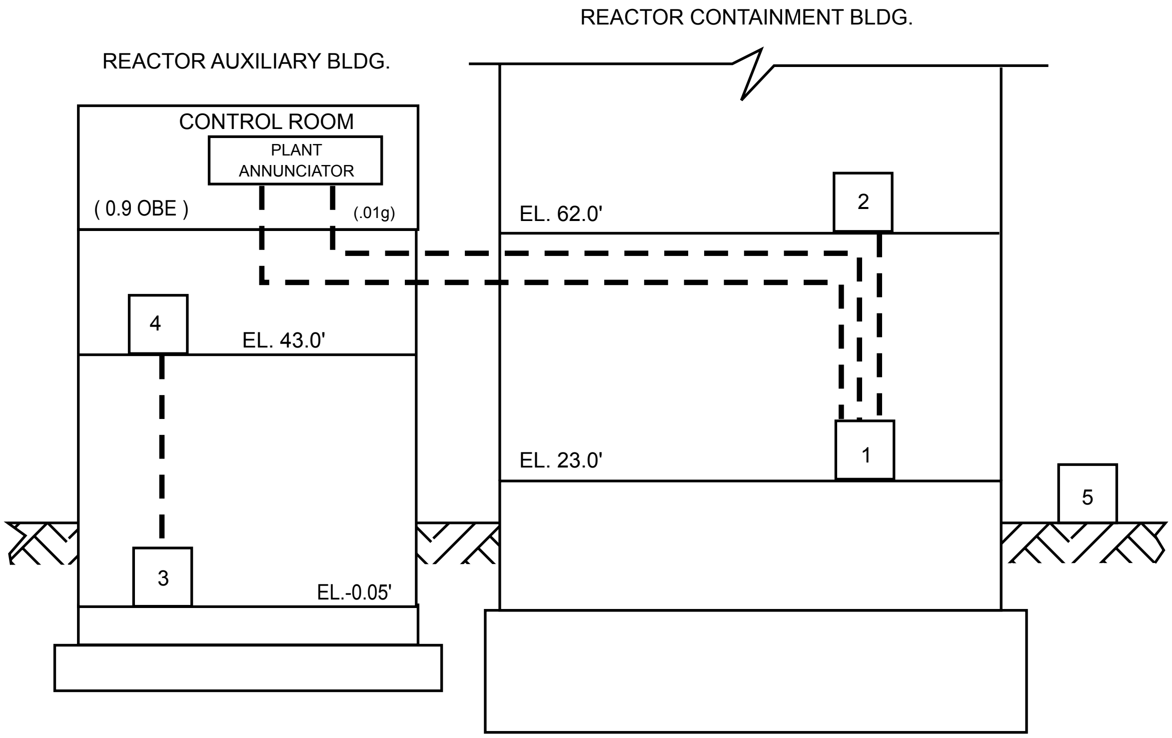
FIGURE 3.7-38



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING CRANE
BRIDGE EARTHQUAKE RESTRAINT

FIGURE 3.7-39



ITEM	TAG NO.	DESCRIPTION
1	SMR-42-1	TRIAXIAL ACCELEROGRAPH - EL. 23.0'
2	SMR-42-2	TRIAXIAL ACCELEROGRAPH - EL. 62.0'
3	SMR-42-3	TRIAXIAL ACCELEROGRAPH - EL. 0.50'
4	SMR-42-4	TRIAXIAL ACCELEROGRAPH - EL. 43.0' EQUIP SUPPORT
5	SMR-42-5	TRIAXIAL ACCELEROGRAPH - FREE FIELD

Florida Power & Light Company
St. Lucie Plant Unit 1

Seismic Monitor Arrangement
Figure 3.7-40

3.8 DESIGN OF STRUCTURES

3.8.1 SEISMIC CLASS I STRUCTURES OTHER THAN CONTAINMENT

3.8.1.1 Design Bases and Description

3.8.1.1.1 Reactor Auxiliary Building

The reactor auxiliary building is a reinforced concrete structure with cast-in-place concrete exterior walls. The interior floor construction is of beam and girder construction supported by reinforced concrete columns. All interior shielding walls are either solid concrete block of reinforced construction, or reinforced concrete.

The reactor auxiliary building is a seismic Class I structure and houses the waste treatment facilities, engineered safety features, switchgear, laboratories, offices, and control room. It further provides protection to the cable and piping penetration areas of the reactor building. The building exterior walls, floors and interior partitions are designed to provide plant personnel with the necessary biological radiation shielding and protect the equipment inside from the effects of adverse atmospheric conditions including tornado and hurricane, winds, temperature, missiles and external flooding. The reactor auxiliary building general arrangement is shown on Figures 1.2-12 through 1.2-17.

3.8.1.1.2 Fuel Handling Building and Cask Handling Crane Support Structure

The fuel handling building is a reinforced concrete structure enclosing the spent fuel pool, cask pit and support equipment. The spent fuel pool and contiguous cask pit are a single cast-in-place steel-lined reinforced concrete tank structure that provides space for underwater storage of spent fuel, a spent fuel cask, and miscellaneous items. The remainder of the fuel handling building consists of cast-in-place concrete exterior walls with interior walls which are reinforced concrete construction. The floors and roof are of beam and girder construction supported by columns. The fuel handling building houses heating and ventilating equipment, the fuel pool heat exchanger, fuel pool filter, fuel pool cooling pumps, and fuel pool purification pump. In addition, the fuel handling building provides an area for cask loading and space for the storage of new fuel and a decontamination area for the spent fuel cask and miscellaneous equipment. Attached to the north outside wall of the fuel handling building, the Cask Handling Facility provides an area where spent fuel casks are prepared for dry storage. The casks are drained, sealed, and inerted with helium before being transferred to the Horizontal Storage Module (HSM) for storage.

The outdoor cask handling crane located above the FHB roof is capable of hoisting a spent fuel cask through a roof opening that is normally covered by an L-shaped door directly above the cask pit in the northeast corner of the FHB. The crane's external runways and steel frame structure are supported by the FHB roof and east exterior wall, as well as by columns on concrete foundations at the grade elevation.

Both the fuel handling building and cask handling crane support structure are designed as seismic Class I structures. The building exterior walls, floors, and interior partitions are designed to provide plant personnel with the necessary biological radiation shielding and protect the equipment inside from the effects of adverse atmospheric conditions including tornado and hurricane winds, high temperature, external missiles, and flooding. The fuel handling building general arrangement is shown on Figures 1.2-18 and 1.2-19.

3.8.1.1.3 Diesel Generator Building

The diesel generator building is a reinforced concrete structure housing duplicate diesel generator units, each separated from the other by a reinforced concrete wall. The fuel supply is maintained in tanks located a short distance from the building.

The diesel generator building consists of a common reinforced base mat, exterior walls and concrete roof. A single interior wall will separate the duplicate units. The diesel generator sets are supported on a pedestal on the base mat.

The diesel generator building is designed as a seismic Class I structure. The building exterior walls and roof are designed to protect the equipment inside from the effects of adverse atmospheric conditions including tornado and hurricane winds, missiles and flooding. The structure openings are also designed to protect against tornado debris (see Appendix 3F).

3.8.1.1.4 Intake Structure

The intake structure is a reinforced concrete structure containing the circulating water pumps and the intake cooling water pumps. The structure consists of a base mat founded wholly in Class I fill, exterior walls braced internally to the bay walls and an operating deck. The mechanical valve pit is located to the north. Water enters the structure through four submerged openings, passes through traveling screens before entering the rear of the structure where the various pumps are located. The intake structure is serviced by a 45 ton capacity bridge crane.

Seismic Class I retaining walls to the north and south of the intake structure provide support for the fill in their respective areas in addition to providing foundations for the bridge crane.

The intake structure is designed as a seismic Class I structure. The structure, with associated retaining walls, provides support for the intake cooling water pumps and piping and bridge crane. The structure is designed to withstand seismic, tornado, missile and hurricane loadings and flooding.

3.8.1.1.5 Ultimate Heat Sink Dam (Barrier Wall)

The dam is a reinforced concrete buttressed retaining wall which extends across the ultimate heat sink canal connecting Big Mud Creek to the intake canal. Its function is to separate the waters of Big Mud Creek from the intake canal during normal operation, and through valved openings, provides an alternative source of cooling water in the unlikely event that the ocean intake becomes unavailable.

The dam is a Seismic Class I structure designed to withstand OBE, DBE, Tornado (components enclosed by 2' concrete), missiles and PMH loadings. The foundation of the dam is at elevation -20.0 and is resting on approximately 6 feet of compacted Class I backfill. A steel sheet piling cut off wall driven to bottom elevation -34.0 prevents any underseepage.

The Barrier wall is to be constructed in the dry with the excavation protected by earthen dikes and de-watered by a well point system.

The seismic analysis of the Barrier wall has been made with a three lumped mass spring model similar to the analysis used for other structures as discussed in Section 3.7 but using methodology specified in Section 3.8.1.7.5.

3.8.1.2 Design Codes

The reactor auxiliary building, fuel handling building, diesel generator building and intake structure are designed in accordance with the "ACI Standard Building Code Requirements for Reinforced Concrete" ACI 318-63 Part IVB, Ultimate Strength Design. No concrete structure was designed to ACI 318-71 since this code did not exist at the time the plant construction permit was granted.

Design, fabrication and erection of structural steel is in accordance with AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," 1969 and the "Code of Standard Practices for Steel Buildings and Bridges", July 1970. All steel shall be ASTM A36, unless noted.

3.8.1.3 Materials, Quality Control and Special Construction Techniques

The materials used in all reinforced concrete structures met the following requirements: At start of construction and, except as noted within the Ebasco Concrete Specification No. 8770.473, as these and all other applicable ASTM Standards were revised, the latest revisions were followed.

a) Concrete

The aggregate as a minimum conforms to all the requirements of "Specification for Concrete Aggregates" ASTM-C33-67. The hardness, weight, strength, durability and reactivity as well as gradation must the satisfactory and fall within the limits established for a good grade of concrete. Mixing water is required to pass ACI 318-63 requirements.

Concrete test mixes are determined using the same materials being furnished for the job. Test cylinders are prepared and tested with results studied when 7 and 28 days old.

Air entraining agents ate used. Retarding agents are added when field construction conditions make it desirable. Air entraining admixtures conform to ASTM-C 260-68. Retarding agents conform to ASTM-C 494-68.
In no case is calcium chloride used in admixture.

b) Reinforcing Steel

All required reinforcing is new billet steel accordance with ASTM A615-68. Mill tests results are obtained from the reinforcing steel supplier for each heat of steel to show proof that the reinforcing steel has the specified composition, strength and ductility. All reinforcing steel is shipped to, the job in bundles bearing a tag identifying its size, grade and code number keyed to heat numbers. This information is verified by certified mill test reports which accompany each shipment of reinforcing steel. Bars No. 11 and smaller are lap spliced in accordance with ACI-318-71, section 805.

Cadweld splices are not used if alternative design or construction is available. If cadweld splices are needed, all-such splices are made in accordance with the requirements of Regulatory Guide 1.10. Approximately 550 mechanical splices (cadwelds) were used for the vertical reinforcing of the steam generator shield walls at elevation + 62.0 in the reactor building.

c) Welding Materials

A variety of filler metals were used in welding the various base metals required for plant construction. The weld rods used for each base metal, and the welding process used are listed below.

BASE METAL	FILLER METAL	WELDING PROCESS
A516 Gr7O-A300 (ASME P- 1)	E7018	Shielded Metal Arc
A516 Gr 70-A300 (ASME P-1)	RACO Med. MnMo F74 Flux E7018 Handpass	Submerged Arc Shielded Metal Arc
A516 Gr 70-A300 (ASME P-1)	E70 T-1 CO_2 Gas With E7018 Handpass	Gas Metal Arc Shielded Metal Arc
A516 Gr 70-A300 (ASME P-1)	EK12K & F72 Flux E7018 Handpass	Submerged Arc Shielded Metal Arc
INCO 182 (ASME P-43) Overlay on A516 Gr 70 Welded together	E NiCr Fe-3	Shielded Metal Arc
SB168 (Inconel) (ASME P-43) to A516 Gr 70	E NiCr Fe-3	Shielded Metal Arc
SA333 Gr 1 to SA312 type 304	E309	Shielded Metal Arc
A516 Gr 70-A300 (ASME P-1)	E70T-G	Gas Metal Arc
A516 Gr 70-A300 (ASME P-1)	EM12k & F72 Flux	Submerged Arc
A516 Gr 70-A300 (ASME P-1)	E70 T-1/CO_2 Gas E7018	Gas Metal Arc Shielded Metal Arc
A240 Tp 304 (ASME P-8)	E308L	Shielded Metal Arc
A333 Gr 1 (ASME P-1)	E70S-2 E7018	Gas Tungsten Arc Shielded Metal Arc
A312 Tp 304 (ASME P-8)	ER308L	Gas Tungsten Arc
A516 Gr 70-A300 (ASHE P-1)	E7018 EK12k	Shielded Metal Arc Submerged Arc
A516 Gr 70-A300 (ASME P-1)	E70T-1 EM12k	Gas Metal Arc Submerged Arc
A333 Gr 1 (ASME P-1)	E7018	Shielded Metal Arc

3.8.1.4 Design Loads

Loadings included in the design of the reactor auxiliary building, fuel handling building, diesel generator building and intake structure are as follows:

a) Structural Dead Load (D)

Dead Load consists of the dead weight of the concrete structure, superstructure, walls and miscellaneous building items within the building.

Specific weights for dead load calculations are as follows:

1 -- Concrete	:	138 lb/cu ft
2 -- Reinforcing Steel	:	as specified in ASTM A615
3 -- Structural Steel	:	489 lb/cu ft

b) Live Load (L)

Live loads are set to assure a structure sufficiently strong under normal operation to support equipment, random temporary load conditions for maintenance, and to assure structural adequacy for normal or construction loading.

c) Equipment Load (L')

Equipment Load is the load imposed by the equipment at rest.

d) Wind (H_u)

194 mph wind ASCE Paper No. 3269

e) Tornado (W)

The simultaneous occurrence of the following:

1 - Wind: 300 mph wind uniform with height distribution in accordance with ASCE Paper No. 3269

2 - Pressure: 2.25 psi equivalent internal pressure (nonimpact).

f) Operating Pipe Anchor Load (A)

6000 lb horizontal thrust applied at any point on a vertical exterior wall or on floors, walls, and roof of pipe chase.

g) Pipe Accident Load (R)

1 - Pipe rupture loads include a dynamic factor of 2.0 for piping 2 1/2 inches or more and 3.0 for piping 2 inches or less.

2 - Pipe restraint 30,000 lb in critical areas of reactor auxiliary building (RAB).

3 - Pipe Tunnel: 1 psi pressure differential. RAB pipe tunnel only.

4 - Piping Penetration Area: 1 psi pressure differential. RAB.

5 - Pipe Anchor: 8000 lb per anchor point

h) Design Basis Earthquake (DBE)

DBE Base Ground Acceleration 0.10 g

i) Operating Basis Earthquake (OBE)

OBE Base Ground Acceleration 0.05 g

j) Buoyancy (B)

Displacement of groundwater.

k) Thermal (T)

The load induced by normal thermal gradients existing across the walls between the building interior and the ambient external environment. The conditions are:

1 - Summer

a - Interior sustained air temperature: 74-104 F

b - Exterior sustained air temperature: 93 F

c - Exterior sustained soil temperature: 70 F

2 - Winter

a - Interior sustained air temperature: 50-80 F

b - Exterior sustained air temperature: 32 F

c - Exterior sustained soil temperature: 70 F

For all cases, an "as constructed" concrete temperature is assumed at 70 F.

l) External Missiles (M)

External structures are designed to withstand without perforation the impact of high velocity external missiles as might occur during the passage of a tornado. Representative missiles considered in the design are discussed in Section 3.5.2.2.

m) Earth (S)

1 - Unit weight dry earth - 105 pcf

2 - Unit weight saturated earth - 125 pcf

3 - Unit weight submerged earth - 60 pcf

4 - Horizontal earth pressure

a - Active Ka - 0.3

b - Passive Kp - 3.0

5 - Under earthquake, earth load shall be considered as maximum passive earth pressure.

3.8.1.5 Load Combinations

The following load combinations are used in the design of the reactor auxiliary building, fuel handling building, diesel generator building and intake structure:

a) Normal Operation

U = 1.5 (D+T+S) + 1.8 (L+A) +1.0B

b) Operating Basis Earthquake

U = 1.25 (D+T+L' + 0.2L+A+OBE+S) +1.0B

U = 0.9 (D+T+L'+S) +1.1OBE + 1.0B

C) Design Hurricane

U = 1.25 (D+T+L' +0.2L + A +H_u + S) + 1.0B

U = 0.9 (D+T+L'+S) + 1.1 (H_u + B)

d) Design Basis Earthquake

U = (1.0 ± 0.05) (D+T+L' + 0.2L) + 1.0 (A+R+DBE+ S+B)

Where uplift is critical, the live load is omitted.

e) Tornado

U = (1.0 ± 0.05) (D+T+L' +0.2L) +1.0 (A+W+M+S+B)

Where uplift is critical, the live load is omitted.

f) Accident

U = (1.0 ± 0.05) (D+T+L'+O.2L+A+S) + 1.0R + 1.0B

3.8.1.6 Design Stress Limits

Using the factored load combinations as defined in Section 3.8.1.5, the various components have the required load capacity when the stresses in them do not exceed the yield strength of the materials used.

The yield capacity of all load carrying structural elements is reduced by a yield capacity reduction factor (Φ) as given below. This factor will provide for the possibility that small adverse variations in material strengths, workmanship, dimensions, control, and degree of supervision while individually within required tolerance and the limits of good practice, occasionally may combine to result in undercapacity. The yield reduction factors are as follows:

a) Flexure Φ = 0.90

b) Flexure at lapped reinforcing bars Φ = 0.85

c) Diagonal tension Φ = 0.85

d) Bond and anchorage = 0.85

e) Spirally reinforced compression members Φ = 0.75

f) Axial tension Φ = 0.90

g) Tied compression members Φ = 0.80

h) Axial tension at lapped reinforcing bars Φ = 0.85

i) Axial tension at mechanically spliced reinforcing bars Φ = 0.90

The capacities of all sections are computed in accordance with ACI 318-63 Part IV-B Ultimate Strength Design. For tornado loadings, a concrete stress intensity of 0.75 f'c is used at ultimate strength instead of 0.85 f'c specified in ACI 318-63.

All reinforced concrete structures including Class I structures were designed using the ultimate strength design methods of ACI 318-63. The maximum concrete strain at the extreme compression fiber was limited to 0.003 under factored loads. All deflections were controlled in accordance with Section 1507 of the above mentioned codes.

Steel framing and anchorages were designed to resist increased pipe break loads and thermal loads, where applicable. Allowable steel stress was taken as 90 percent of yield stress.

The shield structure and the reactor interior structure were analyzed by computer programs assuming linear and elastic behavior of the concrete section. The computer results indicated that no permanent deformation of structures will take place under all factored loads.

3.8.1.7 Design Analysis Methods

The reactor auxiliary building, fuel handling building, intake structure and diesel generator building are analyzed for the load combinations given in Section 3.8.1.5 in accordance with the ultimate strength design (USD) methods of ACI-318-63. Details of the seismic analysis of these structures are given in Section 3.7.2.

The ultimate strength method of the ACI 318-63 code is used for the design of the concrete foundations. All foundations and supports carrying seismic Class I components are seismic Class I.

The specific load combinations used for foundations are:

1.0 (D + T) + 1.0 LOCA + 1.0 DBE

1.0 (D + T) + 1.0 W

The FRAN (FRame ANalysis) program has been used in analyzing the reactor auxiliary building, fuel handling building, intake structure and diesel generator building. FRAN is a computer program which can completely analyze all types of elastic, statically loaded, three dimensional structures having slender members. The member connections can be rigid, semirigid or free (simple beam). Outputs of bending moments, end forces, joint displacements and member distortions can be obtained from FRAN when inputs of loads, coordinates of all joints and members with their structural properties and end conditions are supplied. The structural analysis is based on double precision stiffness matrix manipulations.

3.8.1.7.1 Reactor Auxiliary Building

a) Mat

The foundation mat is designed as a series of two-way slabs between column lines in accordance with ACI 318-63 Appendix A, Design of Two-Way Slabs.

b) Columns

Columns are designed as columns in accordance with ACI 318-63 code, Chapter 19, Combined Axial Compression and Bending, USD.

c) Beams, Girders and Slabs

Beams, girders and slabs are designed as flexural members in accordance with ACI 318-63, Chapters 16, Flexural Computations, USD, and 17, Shear and Diagonal Tension, USD. Slabs are designed in accordance with ACI 318-63, Appendix A, Design of Two-Way Slabs.

d) Exterior Walls

Exterior walls are designed as slabs for tornado and earthquake loadings in accordance with ACI 318-63. They are also designed as shear walls in accordance with ASCE Manual of Engineering Practice No. 42, "Design of Structures to Resist Nuclear Weapons Effects," 1964 edition.

3.8.1.7.2 Fuel Handling Building

a) Mat

The foundation mat is designed as a rigid mat in accordance with flexural and shear stress requirements of ACI 318-63, USD.

b) Walls

Fuel Pool walls are designed as a rectangular water tank meeting flexural and shear stress criteria of ACI 318-63, USD. Other exterior shear walls are designed for shear and moment with the structure taken as a complete structural cross-section resisting design shears. Flexural and shear stresses meet the criteria of ACI 318-63, USD. Individual wall panels are checked as two-way and one-way slabs against tornado loadings in accordance with ACI 318-63.

c) Columns and Beams

Columns and beams are designed in accordance with ACI 318-63, Chapter 19, "Combined Axial Compression and Bending - USD," and flexural and shear stress requirements of ACI 318-63, USD.

d) Frame Analysis

The frame analysis of the structure was completed using IBM Program H20-0340-2, "Analysis of Structures with Prismatic Members in 2 and 3 Dimensions," with either pinned or rigid joints and subjected to concentrated or distributed loads, displacements and temperature effects.

3.8.1.7.3 Intake Structure

a) Mat

The foundation mat is designed as a beam spanning between the walls of the structure. Flexural and shear stresses are determined in accordance with the requirements of ACI 318-63, Chapter 16, "Flexural Computations, USD," and Chapter 17, “Shear and Diagonal Tension USD."

b) Walls

Intake structure walls are designed as flexural members spanning between either bay walls or struts meeting flexural and shear stress requirements of ACI 318-63, USD as in (a).

c) Deck

The deck is designed as a flexural member spanning between bay walls meeting ACI 318-63 as described above.

3.8.1.7.4 Diesel Generator Building

a) Mat

The foundation mat is designed as a rigid mat meeting flexural and shear stress requirements of ACI 318-63, USD.

b) Walls and Roof

The walls and roof are designed as a two-way portal frame using Ebasco Computer Program No. 117, "Rigid Frame Analysis," which uses a matrix analysis method to analyze a planar rigid frame assembly. Walls and roof are then designed to meet flexural and shear stress requirements of ACI 318-63, USD.

3.8.1.7.5 Ultimate Heat Sink Dam (Barrier Walls)

The structural elements shall be designed in accordance with the strength design provisions of ACI 318-71 Code. The stem of the wall is designed as a slab supported on three sides (counterforts on two sides and the bottom slab). The toe and heel slabs are designed as continuous on three sides (counterforts on two sides and continuous under the stem). The counterforts are designed as buttresses if under compression and as ties if under tension for the slab reactions.

The wall stability is calculated about a plane at its base. The procedure followed in analyzing the wall about its base for computation of base pressure, overturning and sliding to determine the horizontal. and vertical forces acting on the structure and to determine the ratio of moments resisting overturning to moments causing overturning, and the ratio of forces resisting sliding to the forces causing sliding. Generalized earth pressure coefficient equations derived from Coulomb's theory have been utilized in the analysis. The total pressure acting on one side of the wall is the sum of the static earth and water pressures plus the dynamic pressures of each due to the horizontal acceleration during an earthquake. Against this pressure acting on the other side of the wall is a total pressure consisting of the sum of the passive pressure due to earth and the active pressure due to water reduced by the dynamic pressures of each due to the horizontal acceleration during an earthquake. The dynamic force of the earth due to earthquake is calculated by finding the total weight of the sliding wedge taking the inertia of it by multiplying by the earthquake acceleration and converting this load to an equivalent inverted triangular load applied to the depth of the wall from zero at the base to a minimum at the ground surface. The dynamic force of the water due to earthquake is calculated by using the Westergaard parabolic equation. This is illustrated in Figure 306 of the report of the "Design of Kentucky Structures Against Earthquake." The dynamic force due to concrete is calculated by a spring model of lumped masses.

For the stability analysis the minimum allowable safety factor under normal operating loading conditions shall be (1) overturning 1.7; (2) sliding 1.5. For extreme (transient) loading the safety factors shall be 1.2 for both overturning and sliding. No load factors shall be used in the stability analyses. Where the analysis shows an uplift condition occurring the area of the foundation in contact with this is greater than 75 percent of the total area. The horizontal and vertical seismic effects are combined by using either square root of the sum of the squares of the maximum values or the maximum of the time history sum.

Table 3.8-4a presents the results of the stability analysis for critical loading combinations. It should be noted that the loading combinations of Section 3.8.4.3.2 of St. Lucie 2 FSAR (Docket number 50-389, dated September 4, 1973) are used for analysis where they are more limiting. The lateral water level loads during normal (F) and earthquake (F') situations have been added to the lateral earth loads in all loading combinations.

3.8.1.7.6 RAB and FHB Interior Masonry Walls

Pursuant to an NRC request, IE Bulletin 80-11 "Masonry Wall Design" (Reference 11), a field inspection program and a design reevaluation program were undertaken to verify the adequacy of the design of the interior masonry walls in the Reactor Auxiliary Building and the Fuel Handling Building.

The criteria used for the reevaluation is as follows. The stresses resulting from the combination of dead load, SSE, and pressure load caused by failed equipment inside the masonry wall compartment must remain below the allowable stresses listed in column "U" in Tables 3.8-4B and 3.8-4C. These are based on ACI 531-79, "Building Code Requirements for Concrete Masonry Structures."

At the completion of the program certain corrective actions were taken thus bringing all the masonry walls into compliance with the criteria.

The procedures for the inspection and verification programs and the results of those programs and corrective actions taken are discussed in the final report. See References 12 and 13.

3.8.1.8 Calculated Results

The loads, moments and stresses have been calculated for each of the design loading combinations given in Section 3.8.1.5 for the reactor auxiliary building, fuel handling building, intake structure and diesel generator building. In all cases the calculated design loads are within the ultimate capacity of the structural members. Tables 3.8-1 through 3.8-4 give a comparison of the calculated values of shear, moment and stress with the ultimate capacity for the principal structural members of the various structures. The calculated values are given for the most severe loading conditions for the particular member.

Below is a typical calculation for the design of a component of a Class I concrete structure. In the example given, Figure 3.8-1a, the critical loading condition is normal operation which was determined after checking all other loading conditions as listed in Section 3.8.1.5.

The loads in this example are determined from the physical arrangement of the structure. The design moments and shears are determined from design tables given in "Reinforced Concrete Design Handbook," ACI Manual SP-3.

The design moments and shears are then used to determine the flexural steel required and the shear stirrup design. The "Ultimate Strength Design Handbook," ACI SP-17, has been used for this design. The combination of bar size and spacing required to fit the steel in a concrete member is such that more steel than is required is often used.

The ultimate capacity of the member is determined using the actual beam properties and analyzing the beam again using the previously mentioned "Ultimate Strength Design Handbook."

DESIGN LOAD DETERMINATION -

Design load combination: Normal Operation

Ultimate load = 1.5 (D + T + S) + 1.8 (L + A) + 1.0B

For interior floor beam T, S, A B ~ 0

$\therefore$ Ultimate load = 1.50 + 1.8L

Live load on floor = 200 psf

Ultimate load on beam

Span KL

Uniform Load:

W live load = $\left[\frac{1}{3} x\,24 + 3 + \frac{1}{3}\ x\ 7\ \right] x\,1.8\,x\,0.2 = 4.8\ \ K/ft$

W dead load = $\left[\frac{1}{3} x\,24\ \ x\,0.35 + 3\,x\,0.58 + \frac{1}{3}\ x\ 7\ \ x\,0.2\ \right] x\,1.5 = 7.5\ \ K/ft$

Concentrated Load:

P live load = $1.8\,x\,0.2\,x\,9\,x\,12\ \ \cdots = 39\,K$

P dead load = $1.5\,x\ \ [9\,x\,0.2 + 2\,x\,0.15\,]\,x\,12 = [1.5\,x\,12\,x\,1.2\,x\,.1] = 60\,K$

Span JK:

Uniform Load:

W live load = $1.8\,x\,0.2\left[\frac{12.5}{2} x \frac{24}{3} + 3\right] = 6.2\,K/ft$

W dead load = $1.5\,\mathrm{x}\left[\ 0.35\,\mathrm{x}\left(\frac{12.5}{2} + \frac{24}{3}\right) + 3\,\mathrm{x}\,0.58\right] = 10.1\,\mathrm{K/ft}$

Use JK as standard span

L = 24'-0" W_e = 6.2 W_t = 16.3

W_d = 10.3

DESIGN

Span 2KL:

Design moment

@ Support = 1160 K-ft

@ Midspan = 690 K-ft

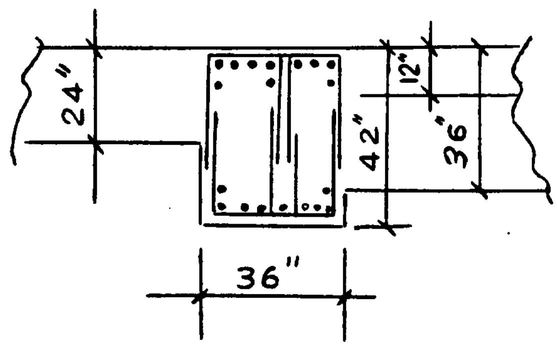
Design shear

@ Support = 285 K

@ span = 0 K

Width of beam = 36" = b
Effective depth = 38" = d
(assumed)

F = 4.33 (From Table 5 of ACI SP17)



@ Support Reinforcement:

$$K_u = \frac{Mu}{F} = \frac{1160}{4.33} = 268$$

$$A_u = 2.81 \text{ (From Table 1.1 of ACI SP17)}$$

$$A_s = \frac{M_u}{A_u d} = \frac{1160}{2.81 \, x \, 38} = 10.9 \, in.^2$$

Provide 10 # 10 bars

@ top = 12.7 in.^{2}

∴ OK

@ Span Reinforcement:

$$K_U = \frac{690}{4.33} = 160 \quad A_u = 2.88$$

$$A_s = \frac{690}{2.88 \, x \, 38} = 6.3 \, in.^2$$

Provide 9 # 9 bars

@ bottom = 9 in.^{2}

∴ OK

Moment multiplier $= W_tL^2 = 163 \times 24^2 = 9400$

$$\frac{\sum K_{Column}}{\sum K_{Beam}} = 2 \quad \frac{W_l}{W_d} = 0.5$$

	L	K	J
a	1.0	1.0	
Table 1A (ACI SP-3)	-.06 + .052 - .095	-.09 + .045 -.086	
Table 2A Conc. Coefficient $\times \frac{P_{ta}}{P_{ta}+W_{ta}}$	-.016 +.014 -.018	-.018	
Table 2A Uniform Coefficient $\times \frac{W_{ta}}{P_{ta}+W_{ta}}$	-.008 +.007 -.01	-.003 +.02	
	-.084 +.073 -.123	-.111 + .045 -.066	
Moment K/ft	-790 + 690 -1160	-1050 + 420 –620	(Design Moments)

Shear Reinf. @ Support

Per. shearing stress $V_c = 2 f_c$

Shear carried by concrete V_c = V_c x b x d

$= \frac{93}{1000}$ x 36 x 38

= 127 kips

Shear to be taken by reinforcement = 285 - 127

= 158 K

Assuming 9" spacing

$$A_u = \frac{V_u \; S}{f_u \; d} = \frac{158 \text{ x } 9}{.9 \text{ x } 40 \text{ x } 38}$$ = 1.04 in.^{2}

Provide 4 # 6 bars
@ 9"
A_s = 1.76 in.^{2}

∴ OK

Shear Reinf. @ Span:

No. reinforcement required

Provide A_s = 1.76 in.^{2}

ULTIMATE CAPACITY CHECK -

36" x 42" (deep) beam with 10 # 10 bars @ top and
9 # 9 bars midspan
@ support

Ultimate Moment Capacity:

$$P = \frac{12.7}{36 \text{ x } 38} \quad = \quad .0092$$

$$A_u = 2.78 \qquad M_u = A_u \; A_s \; d$$

M_u = 12.7 x 2.78 x 38 = 1340 K-ft 1160 K-ff ∴0 K

Shear @ face of support = (99 + 12.3 x 12) 1.15 = 285 K

Shear capacity @ support

$$\frac{1.76 \times 36 \times 38}{9} + \frac{36 \times 38 \times 93}{1000} = 390K$$

$\therefore$ OK

Ultimate moment capacity @ midspan

$$p = \frac{9}{(36 \times 38)} = .0666 \quad A_u \quad 2.84$$

$$M_u = 9 \times 2.84 \times 38 = 975 \text{ K/ft} \quad 690 \text{ K/ft} \quad \therefore \text{ OK}$$

3.8.1.9 Structural Pre-Operational Testing and Inspection

The same structural testing and inspection was performed for all seismic Class I reinforced concrete structures. The discussion given in Section 3.8.2.2.12 for the shield building applies also for the reactor auxiliary building, fuel handling building, intake structure and diesel generator building.

3.8.2 CONTAINMENT STRUCTURE

3.8.2.1 Containment Vessel

3.8.2.1.1 Description

The containment vessel is a low leakage steel shell, including all its penetrations, designed to confine the radioactive materials that could be released by accidental loss of integrity of the reactor coolant pressure boundary. Physically, the containment vessel is a right circular cylinder (2 inch thick), as shown on Figure 3.8-1, with hemispherical dome (1 inch thick) and ellipsoidal bottom (2 inch thick) which houses the reactor pressure vessel, the reactor coolant piping and pumps, the steam generators, the primary coolant pressurizer and pressurizer quench tank, and other branch connections of the reactor coolant system including the safety injection tanks. The containment vessel penetrations include a construction hatch (Figure 3.8-1a), a maintenance hatch, a personnel air lock, an escape lock and various sized penetration nozzles and are described further in Section 3.8.2.1.10 The containment vessel is also equipped with a walkway, access ladder and a circular crane girder with a crane rail attached to the shell of the vessel. The containment vessel is enclosed by the reinforced concrete shield building described in subsection 3.8.2.2.

3.8.2.1.2 Design Conditions

The functional design bases for the containment vessel are given in Section 6.2. The structural design conditions are:

a) Pressure:

Maximum internal pressure	= 44 psig
Design internal pressure	= 39.6 psig
Maximum external to internal pressure differential	= 0.70 psig

b) Temperature:

Coincident with Design and Maximum pressure	= 264°F*
Operating	= 120°F
Minimum service	= 30°F

c) Wind Loads (considered during construction):

Height Above Grade (Ft.)	Wind Load (PSF)
0 - 30	18
30 - 49	24
50 - 99	30
Above 100	36

(Wind pressures include the reduction for the circular shape of the vessel)

*Design pressure and temperature as included in the purchase specification maximum calculated LOCA pressure and temperature are 38.4 psig and 259°F, respectively.

d) Seismic Loads:

Vertical Earthquake	0.075 g for OBE 0.15 g for DBE

Horizontal Earthquake

Lateral forces equal to seismic coefficients shown on Figure 3.8-2 and Figure 3.8-3 (OBE and DBE) multiplied by the gravity loads.

The design for earthquake includes the seismic effects due to the inertia of the mass of the air locks and equipment hatches and the effects of the air locks vibrating as independent systems.

e) Gravity Loads (include but are not limited to):

Item	Estimated Weight
Vessel shell & appurtances	= 7,880,000 lb.
Penetrations	= 130,000 lb.
Equipment hatch	= 360,000 lb.
Maintenance hatch	= 70,000 lb.
Personnel lock	= 58,200 lb.
Escape lock	= 24,500 lb.
Ventilation duct	= 35,200 lb.
Crane girder rail & girder	= 471,300 lb.
Trolley	= 128,200 lb.

f) Live Loads (include but are not limited to):

Item	Load
Weight of Contained Test Air	= 906,718 lb.
Crane Operating Live Load	= 175 tons
Impact	= 26.25 tons
Air Locks	= 150 psf
Platforms on Dome	= 50 psf
Access Ladder	= 500 lb.
Maintenance Hatch	= 50 tons

g) Occasional Loads

In addition, occasional loads (i.e., loads due to water and steam hammer) are also included in the evaluation of main steam and feedwater piping penetrations.

3.8.2.1.3 Design Leakage Rate

The containment vessel including penetrations is designed to limit leakage to 0.5 percent volume per day at the design internal pressure of 39.6 psig. Containment leakage rate testing is discussed in Section 6.2.1.4.

3.8.2.1.4 Codes

The design, fabrication, inspection and testing of the containment vessel complies with the requirements of the ASME Boiler and Pressure Vessel Code, Section II Materials; Section III, including all addenda through winter of 1968, Nuclear Vessels, Subsection B "Requirements for Class B Vessels;" Section VIII "Unfired Pressure Vessels," and Section IX "Welding Qualifications."

The containment vessel is code stamped for pressures of both 44 psig and 39.6 psig in accordance with Paragraph N-1500 of Section III of the ASME Boiler and Pressure Vessel Code.

The design internal pressure for the containment vessel is specified in accordance with the provisions of Section III of the ASME Boiler & Pressure Vessel Code. The design requirements for Class B vessels are contained in Article 13 of Section III.

The containment vessel was pressure tested in accordance with the rules of ASME Boiler & Pressure Vessel Code, Section VIII UG-100 and Section III N-1314 (d). The maximum test pressure was 1.25 times the design internal pressure (1.25 x 39.6 = 49.5 psig).

The design of supports and bracing and similar structures not within the scope of the ASME Code conform to the requirements of American Institute of Steel Construction (AISC) Specifications, sixth edition.

The containment vessel design and construction also meets all the requirements of state and local building codes.

3.8.2.1.5 Materials

The materials used in the containment vessel are listed on Table 3.8-5. The containment vessel and the equipment hatches and personnel air locks are fabricated of ASME-SA 516 Grade 70 firebox quality steel plate made to SA 300 requirements except that impact test requirements are as specified in the ASME Boiler & Pressure Vessel Code, Section III, N-1211 (a) for a minimum service temperature 30° F.

Penetrations which are integral parts of the containment vessel are of ASTM SA-333 Grade 1 or ASME SB-166 or SB 167 or SB 168.

Charpy V-Notch specimens (ASTM A 370 Type A) used for impact testing of all product forms were in accordance with the requirements of the ASME Boiler & Pressure Vessel Code, Section III N-330.

All ferritic material in the fabrication of the containment vessel has a nil ductility transition temperature of zero degrees maximum when tested
in accordance with the appropriate specification of the material.

During reactor operation, or pressure or leak rate testing the containment vessel metal temperature will be maintained above 30 F.

During the erection of the containment vessel, it was supported by twenty-four temporary steel pipe column assemblies welded directly to the vessel shell. The temporary supports were removed after the containment vessel was completely constructed, post weld heat treated, pressure and leak tested in accordance with the applicable requirements of the ASME Code to demonstrate its integrity and leak tightness, and a portion of the permanent base foundation had been placed. The supports were cut not closer than 1/4 in. from the surface of the shell plate and the remaining support material and welds was removed by chipping and grinding smooth with the shell face. Actual removal of the supports had not begun until the internal concrete had been placed to elevation +7.0 and the external concrete had been placed to elevation +1.0 (see Figure 3.8-3a). In addition, the space between the temporary supports, which is about 8 ft. wide, was concreted to elevation +10.0. The space left after removal of the temporary supports was concreted and the remainder of the sequence was made as discussed below.

A placing and grouting procedure was used to fill void areas beneath the containment vessel. The placing and grouting procedure results in a continuous support of the vessel.

Concrete placements were made according to the plan and sections shown in Figure 3.8-3a. The placing sequence is as follows:

1. Install concrete pour 1 inside vessel
2. Place pour No. 2 under vessel thru holes in form
3. Place pour 2A between initial and final set of adjacent concrete
4. Remove forms around pour No. 2
5. Place pour No. 3 to 6 in sequence as shown on plan
6. Place pours 3A to 6A same as item No. 3
7. After concrete has hardened, remove forms and remove plastic tubes
8. Perform grouting operation
9. Placement of concrete for pour 7 thru 16 to follow in proper sequence as shown on plan
10. Place pours 8A to 16A same as item No. 3 and grouting operation to resume
11. Placement of concrete pours 17 to 32 to follow. No grouting required

In preparation for the grouting operation, the tube voids are first blown out with compressed air to determine that they are free and clear. Next, the interface between the underside of the vessel and the concrete is completely sealed. At the same time, pipe nipples are grouted into place on the end of each grout trough. Each nipple is connected to a tee fitting which allows for installation of a pressure gauge (0 to 20 lbs.) and a check valve for control of pumped grout.

The two components of the epoxy grout were mixed in small batches to prevent their setting before the batch was completely used. Hand pumps were then connected to the pipe nipples and grout introduced to the tube troughs. Pumping continued until a grout return was observed on the other end of the grout tube. The open end was then plugged and grouting continued to allow it to spread into any shrinkage area. The operation was maintained until leakage was observed through the mortar plug around the area being grouted.

The next set of tubes was grouted upon completion of the grouting of one tube and return. This procedure was continued all around the placement until the entire circle was completed. Grouting by this method allows completion of the entire operation before the grout in the first tubes has set beyond the tacky state.

Three two inch cube samples were made of the epoxy during each grouting operation. These are used for testing and record purposes.

3.8.2.1.6 Load Combinations

Various combinations of loads were considered in the design of the containment vessel corresponding to loading conditions during construction, test, normal operation, earthquake and accident conditions. Eleven cases in all were considered as follows:

a) Case 1-Construction at Post Weld Heat Treatment (PWHT)
b) Case 2-Acceptance Test at Ambient Temperature
c) Case 3-Pre-Operation Test at Ambient Temperature
d) Case 4-Normal Operating Condition at Temperature Range of 30°F to 150°F
e) Case 5-Cold Shutdown at Temperature Range of 30°F to 120°F
f) Case 6-LOCA Condition with OBE
g) Case 7-LOCA Condition with DBE
h) Case 8-Pipe Rupture with OBE
i) Case 9-Pipe Rupture with DBE
j) Case 10-Condition with OBE and Thermal plus Seismic loads on piping
k) Case 11-LOCA Condition with DBE with Pressure and Thermal plus Seismic loads on piping

The load combinations for each of the cases are summarized on Table 3.8-6.

3.8.2.1.7 Allowable Stress Criteria

The allowable stresses for each of the load cases are summarized on Table 3.8-7. The allowable stresses were determined by the following methods:

a) <u>Allowable Buckling Stresses for Unstiffened Hemispherical Head</u>

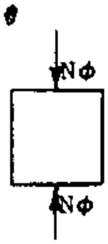
Compressive stress resultants in the top head are compared to the allowable stresses obtained from the paragraphs entitled, "Biaxial Compression-Equal Unit Forces," and "Biaxial Compression-Unequal Unit Forces," of the Welding Research Council Bulletin #69, "Biaxial Stress

Criteria For Large Low-Pressure Tanks." Using these allowables for the spherical dome is based on the assumption that the dome acts as a cylinder with the radius equal to the radius of the dome.

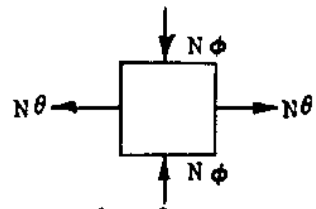
Three cases are considered:

1) For a uniaxial compressive stress resultant and for biaxial unequal tensile and compressive stress resultants.

ie, $N\phi$ (-) Compressive
$N\theta$ 0



or, $N\phi$ (-) Compressive
$N\theta$ (+) Tensile



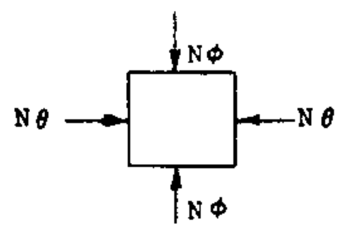
$$N\phi \text{ allowable} = 1.8 \times 10^6 \left(\frac{t}{R}\right)^2$$

Where: t = shell thickness

R - dome radius

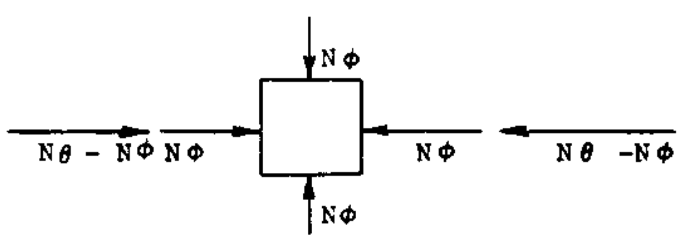
2) For biaxial equal compressive stress resultants

ie, N φ (-) compressive = N θ (-) compressive



$$N\phi \text{ allowable} = 0.9 \times 10^6 \left(\frac{t}{R}\right)^2$$

3) For biaxial unequal stress resultants. This case is treated as the summation of the uniaxial condition with equal stress.

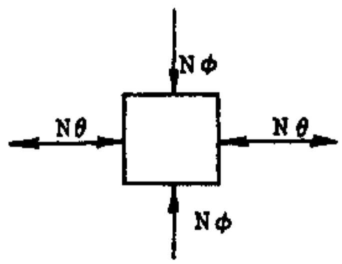


$$\frac{N\theta - N\phi}{1.8 \times 10^6 \left(\frac{t}{R}\right)^2} + \frac{N\phi}{0.9 \times 10^6 \left(\frac{t}{R}\right)^2} \leq 1$$

b) Allowable Buckling Stresses for Cylindrical Vessel

1) Meridonal or Axial Stress

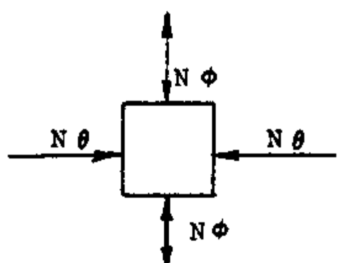
The maximum allowable compressire stress used in the design of cylindrical shells subjected to loadings that produce longitudinal compressive stress is in accordance with Section VIII paragraph UG-23 (b).



Nθ can be tensile or compressive

2) Circumferential Stress

Generally speaking, circumferential compression results from external pressure loading. The criteria of Section VIII paragraph UG-28 is used to analyze circumferential buckling. These rules provide a safety factor of 4.0 against shell buckling.



Nθ can be tensile or compressive

c) Allowable Weld Stresses

All weld metal joining or attaching pressure parts meet specified Charpy V-Notch Impact Test requirements.

1) ASME Allowable Weld Stresses

(a) Full Fusion

Weld allowables per Subsection B of the ASME Code, Section III. Same as parent metal.

(b) Partial Depth Groove Welds

Allowable stress on the effective depth is:

An inspection factor x load factor x Sm of weaker material.
Inspection factor = 0.8
Load factor:

1.0 for load perpendicular to axis of the weld
0.875 for any combination of perpendicular and parallel loads.
0.75 for a load parallel to the axis of the weld

For simplicity, an allowable stress of 0.8 x 0.75 x Sm = 0.6 Sm is used for all partial groove welds except where a higher allowable is required and is permissible.

(c) Fillet Welds

In accordance with ASME Section VIII, paragraph UW-18:

Allowable stress = 0.55 Sm (of weaker material) on min.

leg = 0.55/0.707 Sm on throat = 0.78 Sm on throat

2) AISC Allowable Weld Stresses

(a) Fillet Weld

Allowable Stress = 15,800 psi on throat (AISC 1.5.3.1)

(b) Groove Welds

In accordance with AISC 1.5.3.2.

3.8.2.1.8 Design Analysis Methods

a) <u>Shell Analysis</u>

Stresses in the vessel shell remote from penetrations or other appurtenances are analyzed as described below. Shell stresses adjacent to appurtenances are analyzed along with the appurtenance design.

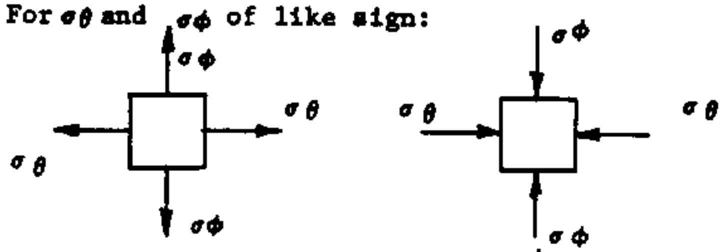
Stress resulting from each specified load condition are calculated separately at critical locations and combined to obtain total meridional and circumferential stresses at each point. Stress intensities are then determined and compared to specified allowable stresses.

In accordance with the maximum shear stress failure criterion and thin shell theory, stress intensities are found as follows:

1) Since shear stress << circumferential or meridional stress, shear stresses are neglected in calculating stress intensities

2) Since radial stress << circumferential or meridional stress, for calculating stress intensities $\sigma_r = 0$

For $\sigma\theta$ and $\sigma\phi$ of like sign:

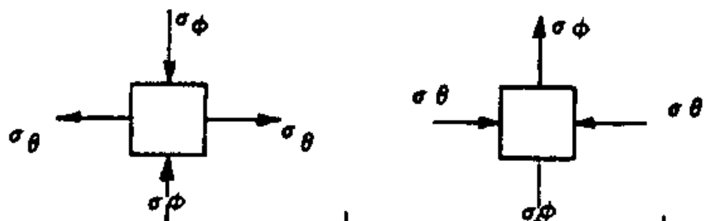


Stress Intensity = Larger of

$$\left|(\sigma\theta) - \sigma_r\right| = \left|(\sigma\theta)\right|$$

$$\left|(\sigma\phi) - \sigma_r\right| = \left|(\sigma\phi)\right|$$

For $\sigma\theta$ and $\sigma\phi$ of unlike sign:



Stress Intensity = $\left|(\sigma\theta) - (\sigma\phi)\right|$

In addition to the stress intensity evaluation, compressive buckling loads are investigated in the construction, normal operating, and accident conditions.

Seismic analysis of the reactor building, including the containment vessel, is discussed in Section 3.7.2.

b) <u>Bottom Head Buckling Analysis</u>

The bottom head is analyzed for loading conditions which could produce buckling. The loading conditions investigated were during PWHT, during test, and during final construction. A portion of the bottom head and the cylindrical shell is modeled as a ring girder to calculate stresses resulting from externally applied loads. Stresses produced by pressure were calculated using Reference 1, and added directly to the stresses produced by external loads.

Loading Conditions:

1) PWHT:

   (a) Vertical bending from column loads

   (b) Horizontal bending from wind shear and load eccentricity

   (c) Torsion from load eccentricity and pin joint friction

2) Final Test:

   (a) Vertical bending from column loads

   (b) Horizontal bending from wind shear and load eccentricity

(c) Torsion from load eccentricity

(d) Internal pressure

3) Final Contruction:

(a) Vertical bending from column loads

(b) Horizontal bending from wind shear and load eccentricity

(c) Torsion from load eccentricity

c) <u>Discontinuity Stresses at Embedment</u>

The analysis is performed using the program on shells of revolution written by A. Kalnins (Reference 1). The program is used widely by the industry, and its results have been found to be in good comparison with other analytical methods (Reference 2).

The analysis is based on the fact that a rotationally symmetric shell may be divided into a number of short segments in the meridional direction and that the stiffness properties of each of these segments can be determined in relation to eight fundamental variables. By enforcing equilibrium and compatibility between each segment and applying boundary conditions, the value of the fundamental variables can be determined for each segment. Values between each segment can then be determined by integration.

The model used for the analysis is as shown on Figure 3.8-4. The vessel is taken to be rigidly fixed at elevation 19 feet, and the model is taken to extend upward to a point remote from any effect of the local discontinuity due to the fixity at the lower boundary or point of embedment. There has been no consideration of the concrete or shell below elevation 19 feet.

The pressure is included all along the model as an internal pressure equal to the design internal pressure of 39.6 psi.

The temperature gradients which are assumed to exist along the shell are as shown on Figure 3.8-5. The cases analyzed are for a steady state condition only, with the shell assumed to be at some ambient temperature before the gradient is applied along the shell. The maximum temperature differential along the shell is taken to be the difference between the ambient temperature and the temperature of the shell above the embedment region at a specified time after the accident.

It is also assumed that the temperature at the inside surface of the shell plate is the same as that at the outside surface, or that there is no temperature gradient across the thickness of the plate.

In accordance with the maximum shear stress failure criterion and thin shell theory, stress intensities are then found as described previously.

d) Air Lock Seismic Analysis

The containment vessel design for earthquake includes the seismic effects of the air locks vibrating as an independent system. The seismic effect of this independent vibration is then added vectorially to all other seismic effects.

In the analysis, the vibration driving force on the air locks is determined by accelerations derived from the response spectra curves, shown on Figure 3.8-6. The vibrating driving forces is considered to be independent of the vibration modes of the composite containment vessel shield building and foundation system.

For analytical purposes the locks are assumed to vibrate in three independent directions as shown on Figure 3.8-7.

Case I will result in forces and moments being applied to the containment in the meridional plane.

Case II will result in forces and moments being applied to the containment in the circumferential plane. Case III will result in a radial thrust being applied to the containment shell.

Once the natural frequency of the lock is calculated for the longitudinal, circumferential, and radial direction of the containment vessel, it is possible to determine the fundamental period and thus the response acceleration to be applied to the lock. The response acceleration is calculated for the insert to shell junction and then applied to the lock loads to find stresses in the shell.

For longitudinal and circumferential direction: (Case I & II)

$K = M/\theta$ Where: K = Spring constant of shell

$\omega = [K/I_o]^{0.5}$
ω = Angular frequency of lock

$\omega = [M/\theta I_o]^{0.5}$ I_o = Mass moment of inertia of lock about point of support 151 on shell

M = Moment at shell

$T = 2\pi/\omega$ (sec)
θ = Unit rotation at shell to insert junction
T = Fundamental period of lock

The spring constant for the longitudinal and circumferential direction is determined by applying a unit deflection (1 radian) at the shell and determining M using Stresses From Radial Loads and External Moments in Cylindrical Pressure Vessels, P.P. Bijlaard, Figure 3 and 7 for the β based on the proper parameters for the junction.

For the radial direction: (Case III)

$K = P/w$	Where: $W =$	Weight of lock plus insert
$T = 2\pi\,[W/Kg]^{0.5}$	$g =$	386.4 in/sec^2
	$P =$	load
$T = 0.32\,[W/K]^{0.5}$	$w =$	unit deflection at shell to insert junction

Stresses in the shell due to the air locks vibrating as an independent system under horizontal and vertical earthquake have been determined by the use of Reference 3.

A horizontal earthquake acting perpendicular to the lock (Case I), will result in meridional shear and moment being applied to the shell.

A horizontal earthquake acting perpendicular to the lock (Case II) results in a circumferential shear and moment being applied to the shell. An earthquake acting parallel to the lock (Case III) subjects the shell to a radial thrust.

Stresses are checked at three locations; at the neck to insert junction, at the insert to shell junction, and at ½ $[Rt]^{0.5}$ from any local stress area.

Stresses are calculated in the insert and in the shell. An equivalent stress intensity is calculated (per maximum shear theory) and compared to the ASME allowables.

e) Penetration Analysis

The penetrations are analyzed for compliance with the ASME Code. Area replacement is calculated using code rules. Welds for nozzles employing partial penetration attachment are analyzed using code rules. Nozzles with specified loads are investigated for pipe wall stresses and for stresses in the vessel shell.

Inserts for penetrations larger than 2 inch pipe size are checked for area replacement in accordance with paragraph N-1310 of ASME Section III. The pipe wall of nozzles with loads specified as thermal plus seismic are analyzed for primary stresses.

Stresses in the vessel shell resulting from loads applied to penetrations are calculated using Reference 3.

Loads are applied in specified combinations on a penetration of interest and on adjacent penetrations that are on cardinal lines of the central penetration within a distance of $2\sqrt{RT}$ (80.0 in). This limit is chosen since the results of this type of analysis are questionable for greater distances. Load combinations include moments acting concurrently with thrust and torsion acting concurrently with shear. Each load is considered reversible for purposes of determining maximum stress intensity.

Pressure produces a complex state of stress in the shell and penetration at their intersection. As a rational means of estimating these stresses paragraph N451 (b) of Section III, though not specifically applicable to Class B vessels, has been used as a guide. This paragraph assumes that in the vicinity of a penetration reinforced in accordance with ASME rules, maximum membrane pressure stress will not exceed 1.0 Sm and the maximum surface stress will not exceed 1.5 Sm.

The loading combinations of thermal plus seismic loads are evaluated using ASME code allowables. ASME Section III states that requirements of paragraph N-414.1 through N-414.4 must be met for allowable stress intensities. It also states that basic values for the allowable stress intensity, Sm, should be those found in Section VIII. Paragraph N-414.4 limits the primary plus secondary stress intensity to 3.0 Sm. An additional requirement of the code states that the local membrane stress intensity due to pressure and mechanical loads be limited to 1.5 Sm. In defining a local stress region, paragraph N-412 of Section III states that the distance over which the total membrane stress intensity exceeds 1.1 Sm may not extend more than $.5\sqrt{RT}$ and may not be closer than $2.5\sqrt{RT}$ to another region where the total membrane stress intensity exceeds 1.1 Sm. R is the mean radius of the vessel and t is the vessel wall thickness.

The loading combinations of thermal plus seismic plus pipe rupture for penetration nozzles are evaluated using an allowable of 0.9 Sy for membrane stress intensities and 1.5 Sy or 1.8 Sm for primary membrane plus primary bending stress intensity.

3.8.2.1.9 Calculated Stresses

Containment vessel stresses have been evaluated for each of the load cases given in Section 3.8.2.1.6. A summary of calculated stresses as compared to allowable stresses for load cases 1 through 9 is presented on Tables 3.8-8, 3.8-9 and 3.8-10 for the ellipsoidal bottom head, cylinder region and hemispherical dome respectively. Load cases 10 and 11 are evaluated for local stresses.

3.8.2.1.10 Penetrations

a) Design Bases

To maintain containment integrity, containment penetrations have the following design characteristics:

1) They are capable of withstanding the maximum pressure which could occur due to the postulated rupture of any pipe inside the containment vessel.

2) They are capable of withstanding the jet forces associated with the flow from a postulated rupture of the pipe In the penetrations or adjacent to it, while still maintaining the integrity of containment.

3) They are capable of accommodating the thermal and mechanical stresses which may be encountered during all modes of operation and test.

b) Electrical Penetrations

Canister or header plate type penetration assemblies are used for all electrical conductors for the continuation of electrical circuits through the containment vessel, the annulus and the shield building. Sufficient cable slack is provided in the annulus to allow for differential expansion between the containment vessel and the shield building. Cable protection sleeves are provided to give support and protection to the cables in the annular space.

The primary containment penetration is inserted in a containment vessel nozzle and is field welded inside the steel vessel to form the sealing weld. The secondary seal is inserted in a nozzle embedded in the concrete shell of the shield building aligned with the containment vessel nozzle. The secondary seal is field welded to the nozzle in the shield building.

The primary containment penetrations feature hermetic cable sealing achieved by a ceramic, glass or high temperature thermoplastic material bonding to a metal flange. The flange is welded to a header plate or secured by screw threads and a ferrule assembly to a header plate, which in turn is welded to the penetration nozzle. The secondary seal is achieved by either epoxy resin or thermoplastic material forming a continuous seal between the metal canister pipe and all conductors. All penetration assemblies are provided with means to pressurize the primary canisters for monitoring of leakage rates.

As shown in Figure 3.8-8, two structurally different types of canisters are used as follows:

1) Type 1 - 15 kv Medium Voltage Power Penetration Assembly

The primary seal canister is constructed from 14 inch diameter schedule 30 seamless carbon steel pipe per ASME SA-106 grade B. Copper conductors

(500 MCM) are brazed into the 15 kv ceramic bushings which are hermetically sealed in the header plates. The conductors in the secondary seal penetration are brought out as pigtails.

2) Type II - 600 v Penetration Assembly

This type of penetration assembly is designed with modifications depending upon the intended service:

(a) For 600 v power service for 500 MCM and 4/0 AWG circuits the primary seal canister is constructed from 10 in. diameter schedule 20 ASME SA-106 grade B carbon steel pipe. The copper conductors are brazed into the 600 v ceramic bushings which are hermetically sealed in the header plates. The conductors in the secondary seal penetration are brought out as pigtails.

(b) For 600 v power and non-shielded control service for 1/0 AWG, #4 AWG, #8 AWG, #12 AWG, #16 AWG. Same as (a) for some, except that the copper conductors in the primary seal are brazed to a sealing tube which in turn is bonded to the glass seal. Others utilize the design in (d) (below) with the thermoplastic seal.

(c) For shielded control, thermocouple and co-axial cables same as (b), except that the conductors are bonded directly to the glass seal.

(d) For 600 volt power control and instrumentation (C3, C6, and D10), the primary and secondary seal utilizes a header plate design. The conductors are sealed in a stainless steel tube by a high temperature thermoplastic material. The stainless steel tubes are secured to the header plate by screw threads and a ferrule assembly.

Each canister is sealed and tested at the factory for leakage. The only seals that are made in the field are the welds mounting the canisters into the nozzles.

The primary seal penetration assemblies are designed, fabricated and tested in accordance with IEEE-317 April 1971 "Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations."

Electrical penetration E-4 has been modified for use as an outage services penetration. The modification includes a blind flange which is removed for outage service and is equipped with Local Leak Rate Test (LLRT) connection for testing the penetration's integrity when the flange is reconnected after an outage.

A steel plate barrier, shown on Figure 3.8-8AA, is erected inside the containment in the electrical system penetrations. The design criteria for this barrier is as follows:

(a) Design Loads:

The following loads have been considered in the design of the barrier:

D = dead load of the barrier itself

P_R = steam/water jet forces and/or pipe whip reactions resulting from a ruptured pipe. A dynamic load factor of 2.0 had been used in the design of the barrier to account for the dynamic nature of the load.

OBE = operating basis earthquake load

DBE = design basis earthquake load

(b) Load Combinations

The barrier has been designed to withstand the following load combinations within the allowable stresses specified.

Load Combination	Allowable Stress
1) D + PR	90 percent of material yield strength
2) D + OBE	Normal AISC working stress
3) D + DBE	90 percent of material yield strength.

c) Piping Penetrations

All penetrations listed in Table 6.2-16, except the vacuum breakers, penetrate the shield building as well as the containment vessel. Both the containment vessel and shield building are provided with capped spare penetrations for possible future requirements.

All process lines traverse the boundary between the inside of the containment vessel and the outside of the shield building by means of piping penetration assemblies made up of several elements. Three general types of piping penetration assemblies are provided;

Type I - Those which must accommodate considerable thermal movements (hot penetrations).

Type II - Those which are not required to accommodate thermal movements (cold penetrations);

Type III - Those which must accommodate moderate thermal movements (semi-hot penetrations).

All penetration assemblies consist of a containment vessel penetration nozzle, a process pipe, a shield building penetration sleeve and a shield building bellows seal. In the case of cold penetrations the containment vessel penetration nozzle is an integral part of the process pipe. For hot and semi-hot penetrations, a multiple flued head becomes an integral part of the process pipe and is used to attach a guard pipe and in case of the hot penetrations an expansion joint bellows. The expansion joint bellows is welded to the containment vessel penetration nozzle.

At the terminal of a piping penetration assembly near the shield building a low pressure leakage barrier is provided in the form of a shield building bellows seal. The bellows provides a flexible membrane type closure between the shield building penetration sleeve, which is embedded in the shield building, and the process pipe.

The shield building bellows is designed to withstand a design pressure of 5 psig and provide an adequate leak-tight seal consistent with overall allowable shield building leakage.

All containment vessel penetration nozzles are designed to meet the requirements for Class B vessels under Section III of ASME Boiler and Pressure Vessel Code. In compliance with the code, the operating stresses in a containment vessel penetration nozzle caused by the attached penetration assembly is limited to the allowable values given in the code.

The multi-ply bellows expansion joint in the hot pipe penetration assemblies and the shield building bellows seal for all pipes are designed to accomodate maximum combination of vertical, radial and horizontal differential movements between the containment vessel, the shield building and the piping. This design considers the calculated displacements resulting from earthquake, pressure and temperature and relative building settlement.

Types I and III process pipe penetrations are provided with guard pipes to preclude an energy release to the reactor building annulus due to a rupture in a process line. Figure 3.8-9 illustrates the design of both Types I and III penetrations. Both are similar in that their designs consist of a process line passing through a concentric guard pipe. The process line and guard pipe are both welded to a common flued head at one end of the penetration. At the other end, the guard pipe is either anchored at the containment nozzle (Type III penetrations), or is simply supported off the process line itself (Type I penetrations). The Type I penetrations are anchored at their respective flued head on the main steam trestle.

Piping for penetration assemblies is designed in accordance with ANSI B31.7 Class 2 except for the four safety injection penetrations which are Class 1. Multiple flued heads (Type I) are one piece forgings designed to withstand the maximum design pressure at the design temperature of the process line. They are designed in accordance with ASME Section III, Subsection B and Section VIII requirements. The guard pipe is designed to be within allowable limits for the maximum design pressure at the design temperature of the process line, and is designed so that the maximum stresses in the guard pipe will not be more than the code allowable stresses of the material for design load combinations. The original analysis supporting this design criterion is provided in Appendices 3G1 through 3G5 and is kept for documentation and traceability purposes.

App. 3G1 - Flued Head Calculations for Types I & III Penetrations

App. 3G2 - Type III Containment Piping Penetrations Assemblies

App. 3G3 - Type I Containment Piping Penetration Assemblies

App. 3G4 - Flued Head Stress Report, Containment Piping Penetration Assemblies (Type III)

App. 3G5 - Flued Head Stress Report, Containment Piping Penetrations Assemblies (Type I)

Special attention was placed upon the fabrication and the tests and inspections performed during manufacture. For example,

a) The flued head fitting and that section of the process pipe and guard pipe between the flued head fitting and the nearest weld joint outside the containment vessel were hydrostatically tested in accordance with the requirement of ANSI B31.7 Code for Nuclear Power Piping.

b) Water used for the hydrostatic testing of stainless steel penetration assemblies did not have a chloride content exceeding 20 ppm.

c) Expansion bellows assemblies were tested in accordance with ASME Code, Section VIII, Paragraph UG-99.

d) Each bellows was flexed ten (10) times through its specified axial and lateral deflections. This test was performed prior to hydrostatic tests.

e) All carbon and low alloy steel pipe and fittings that make up the process pipe or the guard pipe were subjected to impact testing. The impact testing conformed to the procedures and configuration for Charpy V-notch specimens, Type A, Figure 11, as specified in ASTM A-370, and was in accordance with the requirements of ASME B&PV Code, Section III, N-330. The specimen temperature during impact testing was 0 F. Impact values for pipe and wrought fittings meets 15 ft lbs. Any material failing to meet this requirement was rejected.

f) All welding procedure qualifications and welder performance qualification were in accordance with the latest edition of Section IX of the ASME Boiler and Pressure Vessel Code in effect. Repair welding of base materials was in accordance with a procedure that assured the highest quality results in joining the base materials. Repaired base materials were heat treated or stress relieved as required in the applicable ASTM Material Specification. The repaired area was examined by the same nondestructive test requirements originally specified for the base metal. Those defects, in other than weld metal, which are more than 3/8 in. deep or 20 percent of the wall thickness, whichever is less were reported prior to repair.

g) All fluid boundary containment forgings of penetration assemblies were 100 percent ultrasonic examined. Whenever practicable the stage at which ultrasonic testing was performed for acceptance was in the finished condition after final heat tratment. The actual stage of the ultrasonic test was submitted for approval. In addition, all external and accessible internal surfaces were examined by either magnetic particle or liquid penetrant methods including final machined surfaces such as weld end preparation.

h) All longitudinal seam welds of primary bellows and pipes other than process pipe were full, 100 percent, radiographed.

i) All process pipe girth and longitudinal seam welds were fully radiographed and examined by either the magnetic particle or liquid penetrate method. All other examinations required by the ASTM specification for the associated process line were performed.

j) Guard pipe joint welds were examined by either a magnetic particle or liquid penetrant method.

k) All fillet welds were examined by liquid penetrant methods.

l) Secondary bellows longitudinal seam welds were examined by liquid penetrant methods.

m) Joint of primary bellows to flued head were fully radiographed. Closure welds for which radiography is impractical were examined by a magnetic particle method for carbon steel or a liquid penetrant method for stainless steel on the root and final weld surface in lieu of radiography.

The tests and inspections described above were applied to complete penetration assemblies procured as part of original plant construction. Replacement component parts of the secondary penetrations (e.g. secondary bellows) have been designed, fabricated and tested in accordance with standard manufacturing design and code requirements to be functionally equivalent to the original design.

The supports for seismic Class I piping are designed to maintain the piping stresses within the allowable stress limits of Table 3.9-3 for design, normal, upset, and emergency and faulted conditions. This is done by the utilization of hangers, restraints, and snubbers in the support system. The locations of the restraints in relation to the penetrations vary with the individual pipe design conditions and configurations. Pipe whip restraints are provided for those systems listed in Section 3.6.1. As described in paragraph 3.6.5, the pipe whip restraints are located such that reaction forces from a rupture cannot cause the formation of a plastic hinge. Further discussion of pipe whip protection can be found in Sections 3.6.1 and 3.9.2.5. Pipe whip restraint locations are shown in the figures of Section 3.6.

It should be noted that for the pipe rupture, the design anchor loads for penetrations with guard pipes are based upon the moment that would be expected to cause a full plastic hinge in the process line combined with the force caused by lateral jet loading.

Figures 3.8-8B & 3.8-8A show the restraints adjacent to main steam Type I penetration 1 and the letdown Type III penetration 26. As mentioned above, the restraint locations are dependent upon the individual line characteristics (i.e., pipe size, schedule, internal pressure, temperature, process fluid, and line configuration.)

Type I penetrations are used on the main steam lines and the main feed water lines (See Figure 3.8-9). Moments, axial forces, and transverse forces in the process line due to operating, seismic, or pipe-rupture loads, including loads attributable to fluid transient events such as steam and water hammer, are transmitted through the flued head to the Main Steam trestle where the penetration assembly is anchored. Axial and torsional loading are transmitted to the trestle via trunnions located on the main steam flued head. On the feedwater penetration the axial and torsional loads are transmitted to the trestle via shear plates and trunnions on the flued head, respectively. Transverse loadings and moments are conveyed to the trestle to two shear plates on the trestle from the flued head hubs. Penetration design precludes the transmission of moments from the flued head to the containment nozzle because of the bellows provided in the primary seal nozzle assembly. Process line loads of sufficient magnitude to displace the guard pipe spacers into the ring spacer will be transmitted into containment which then acts as a restraint. This movement, however, is continually resisted by both the shear plates and trunnions as mentioned above. Additionally, a jet deflector plate is provided such that fluid streams exiting from a postulated process line rupture just inside containment cannot cause over-pressurization of the primary bellows.

A Type III penetration assembly is shown on Figure 3.8-9. The penetration assembly is designed to accommodate all forces and moments due to both thermal expansion and pipe rupture or earthquakes. All process line loads due to the various operating modes are transmitted back to the containment penetration nozzle via the flued head and guard pipe.

Table 3.8-10A summarizes the stress levels within selected representative process and guard pipes for Type I and Type III penetrations:

a) The stresses for seismic loadings were obtained from the vendor's flued head finite element analysis.

    1) The stressed areas reviewed were the process line to flued head weld area and the guard pipe to flued head weld area.

    2) The stresses shown are based upon the maximum stress in the immediate weld area for all design loading conditions.

    3) Stresses encountered in the flued head are specifically defined by the finite element analyses. In Tables 3.8-10B and 3.8-10C, the element receiving the maximum flued head stress is listed and compared to its allowable value. More detailed information concerning determination of these values is available in Appendices 3G4 and 3G5. In summary, the highest stresses in the flued head generally occur in the crotch area of the process line - guard pipe flue. Figures of the finite element model used in the stress analysis of the flued head assemblies are also provided on pages 3G-65 through 3G-70.

b) For both Types I & III penetrations, the highest rupture stresses expected are presented.

1) In the Type III penetration assemblies, the highest stresses occured at the containment vessel nozzle where the penetration is attached.

2) In the Type I hinged guard pipe, the highest stresses were at the assumed point of rupture, the middle of the guard pipe.

3) For both rupture cases, stresses were compared with yield. As pointed out in Appendix 3G3, a 1.5 Sm stress criteria is met. It should be noted that not all eleven Type III penetrations are presented in Table 3.8-10A. However, the four selected are representative of those having the highest energy potential for various process line sizes (Refer to Figure 3.8-9 listings). As presented, penetration 5 is representative of 6, 26 is representative of 27, 36 is representative of 37, 38, 39 and penetration 40 is representative of 64. The only size not sampled is penetration 44 a 3/4 inch unit.

The containment penetration assemblies are attached to both the containment and their associated process pipe. For purposes of design and evaluation thereof, these assemblies are part of the process pipe and comply with appropriate code requirements. With regard to the stress level acceptance criterion, a value of $3S_M$ is appropriate. The basis for $3S_m$ allowables for upset, emergency and faulted conditions is Section III of the 1968 ASME B&PV code to which the flued heads were designed. It defines a $3S_m$ allowable stress intensity limit for combined primary and secondary stresses in Class "A" or "B" components for upset conditions. (See paragraph N-414.4 for Class "A" components and N-1314(a) for Class "B" components.) The stresses shown for flued heads welds include thermal stresses and seismic displacement stresses, which are secondary stresses. Except for fatigue analysis, Class 2 assemblies satisfy the design and material requirements for Class 1 components.

It might be suggested that because of the attachment to containment that some portions of the flued head should be considered part of containment. Accordingly, a 3S stress level acceptance criterion would be appropriate for some portion of the assembly. Where S is taken from Table VCS-23 of ASME Section VIII Division I. Although $3S_m$ is considered the appropriate stress level acceptance criterion, an evaluation with regard to 3S was conducted. For all but one penetration application of the 3S criterion is inconsequential, i.e., the calculated stresses provided heretofore are well below the 3S acceptance criterion. The penetration germane to this discussion is the main steam penetration. The study discussed below demonstrates that this penetration also satisfies the 3S criterion.

As part of the 3S study, the application of design loads was reviewed. Loading tables furnished the manufacturer specified total moments on an assembly, whereas the manufacturer assumed for design and analysis purposes that these total moments were to be applied at both the inboard and outboard ends of the penetrations. Thus, the results provided in Tables 3.8-10A, 10B & 10C and in

Appendices 3G1 to 3G5 are overly conservative, i.e., the margin between calculated stress intensity and the $3S_m$ criterion is larger than indicated therein. With the total moment properly applied the maximum stresses for the penetration in question has been evaluated. The results are that the maximum stress intensity is 41,038 psi versus a 3S value of 52,500. Thus the stress intensities calculated for the specified design loads for all penetrations comply with both the 3Sm and 3S criteria. A discussion of whether a 3S criterion might be more appropriate is therefore somewhat academic.

The design of the penetration assemblies is as follows:

1) Type I Penetrations

A Type I (hot) piping penetration assembly is used where large thermal movements of the process pipe have to be accommodated and where the differential between the normal operating temperature of the fluid carried by a process line and the containment vessel wall temperature would create unacceptable thermal or cyclic stress at the attachment of the vessel penetration nozzle. A Type I penetration is shown an Figure 3.8-9.

Type I penetrations are used on the main steam lines and the main feedwater lines. A hot penetration assembly has a multiple flued head machined from a solid forging to which are welded in sequence a length of process pipe, guard pipe, and a bellows expansion joint. The multiple flued head is welded into, and becomes an integral part of, the process line. The inner flue provides support for the guard pipe and the outer flue provides support for the expansion joint bellows. The length of guard pipe is set so that it extends past the containment vessel penetration nozzle into the vessel. Near the open end of the guard pipe lugs are provided on the process pipe to serve as limit stops for lateral movement to facilitate distribution of pipe rupture loads, in the event of a slot rupture or pipe whip of the process pipe line within the guard pipe. The guard pipe protects the bellows element against a direct steam impingement in case of a process line rupture. In order to prevent deflection of the guard pipe from overstressing the flued head, at the junction of the flued head and guard pipe, a bellows hinge is provided, protected by a liner, to prevent it from being impinged upon.

The expansion joint bellows is attached at one end to the outer flue on the flued head and at the other end to the containment vessel penetration sleeve. The expansion joint is provided with a double layered bellows that has a connection between the bellows for integrity testing.

A bellows located between the shielded building wall and the flued head, seals the penetration where it passes through the concrete shield wall permitting the annulus between the containment and shield building to be maintained at a slight negative pressure. This bellows is a special two-ply bellows element so constructed as to permit a pressure test of the annulus between the plies.

To provide additional assurance of design adequacy for hot penetrations, an independent review of the design of the hot penetration assembly was performed by Ebasco Services, Inc. The procedure consisted of review and approval of the specified design conditions and of the detailed design by an organization which did not take part in the preparation of the specification or in the detailed design. The specification was prepared by the Ebasco mechanical engineering department and the independent review of the specification was performed by Ebasco's nuclear department. Upon approval, the specification was issued to the penetration assembly vendor as the basis for design. The completed design drawings were subsequently reviewed and approved by Ebasco's mechanical engineering department.

2) Type II Penetrations

A Type II penetration assembly is shown on Figure 3.8-10. This type of penetration is provided for pipe lines carrying low temperature (up to 200°F) and low pressure fluids and gases. The principal consideration in this design is the provision of a leak-tight seal between the pipe and the containment vessel.

This is accomplished by use of sleeves welded into the steel containment vessel by the vessel fabricator. The process line is welded directly to a sleeve penetrating the containment vessel. The sleeve and containment shell is designed to carry the forces and moments due to all operating conditions including a pipe rupture.

3) Type III Penetrations

A Type III penetration assembly is shown on Figure 3.8-9.

For moderate pressures and temperatures over 200 F where there is a possibility of a pipe rupture over-pressurizing the annulus, a multiple flued head is used to provide a leak-tight seal between the penetration nozzle and the process pipe.

In case of a rupture of the process pipe in the annulus area, the guard pipe acts to direct the fluid back into the containment vessel, thus preventing overpressurization of the annulus. The penetration is designed to accomodate all forces and moments due to both thermal expansion and pipe rupture.

The following materials are used in the penetration assemblies:

1) Flued head fittings are in accordance with the requirements of: (1) The ASME Boiler and Pressure Vessel Code, Section III, Paragraphs N-310 through N-313 and N-1210; (2) ASTM Material Specifications A182 Grade F304 alloy steel for stainless steel flued heads and ASTM A105 Grade II carbon steel for carbon steel flued heads.

2) Carbon steel pipe is per ASTM A-106, Grade B. For pipe greater than 24 in. NPS ASTM A-155, Grade KC-70 was used.

3) Stainless steel pipe is per ASTM A-312 (seamless only) Grade TP-304. For pipe 10 in. NPS and greater, ASTM A-358, Class 1, Grade TP-304 was used.

4) The expansion joint material is ASTM A-240 Type 316L.

The foregoing materials in all cases are compatible with the material of the process line.

d) Containment Sump Recirculation Suction Lines

A special type of penetration assembly (Type IV) is provided on the suction lines from the containment sump. These lines are used following a LOCA to allow recirculation of containment sump water by the containment spray and HPSI pumps. Special provisions are made on these lines to reduce the possibility of unisolatable leakage of sump water during recirculation. As shown on Figure 3.8-10, each line consists of a double barrier concentric pipe from the sump up to the suction line isolation valve outside the containment. In the event of leakage of the inner process pipe, the outer pipe will serve to contain the sump water and no uncontrollable leakage will occur. The penetration assembly is designed for the differential motion associated with the DBE.

e) Fuel Transfer Penetration

A fuel transfer penetration (Type V) is provided to transport fuel rods between the refueling transfer canal and the spent fuel pool during refueling operations of the reactor. The penetration is shown on Figure 3.8-11 and consists of a 36 in. diameter stainless pipe installed inside a 48 in. pipe. The inner pipe acts as the transfer tube and is fitted with a double gasketed blind flange in the refueling canal and a standard gate valve in the spent fuel pool. This arrangement prevents leakage through the transfer tube in the event of an accident. The outer pipe is welded to the containment vessel and provision is made for testing welds essential to the integrity of containment. Bellows expansion joints are provided on the pipe to compensate for building settlement and differential seismic motion between the reactor building and the fuel handling building.

The bellows expansion joints meet the requirements of ASME Boiler and Pressure Vessel Code, Section III. The fuel transfer tube bellows are designed for a 35 foot head of water. The static head of water is always less than 35 feet.

Bellows design and construction is such that bellows will not deflect more than its designed amount. The bellows is designed to withstand a 60-year lifetime total of 7,000 cycles of expansion and compression due to operating thermal expansion and 200 cycles of differential settlement and seismic motion.

f) Equipment and Personnel Access

Two equipment hatches are provided. These are welded steel assemblies with 28'-0" diameter and 12'-0" diameter clear openings respectively. The 28'-0" diameter hatch cover will be welded back into position upon completion of construction. The design is such that post-weld heat treatment is not required.

The 12'-0" diameter hatch has a double gasketed flanged and bolted cover. Provision is made to pressurize the space between the gaskets to 44 psig.

Two personnel air locks are provided. These are welded steel assembles. Each lock has two double gasketed doors in series. Provision is made to pressurize the space between the gaskets. The doors are mechanically interlocked to ensure that one door cannot be opened until the second door is sealed. Provisions are made for deliberately violating the interlock by the use of special tools and procedures under strict administrative control. Each door is equipped with quick acting valves for equalizing the pressure across the doors. The doors will not be operable unless the pressure is equalized. Pressure equalization is possible from every point at which the associated door can be operated. The valves for the two doors are properly interlocked so that only one valve can be opened at one time, and only when the opposite door is closed and sealed. Each door is designed so that with the other door open, it will withstand and seal against design and testing procedures of the containment vessel. There is visual indication outside each door showing whether the opposite door is open or closed and whether its valve is open or closed. In addition, limit switches are provided to indicate remotely whether doors are open or closed. Control room annunciation is provided for indication of the Personnel Airlock. Status of the Emergency Escape Air Lock is provided on the security display panel. Provision is made outside each door for remotely closing and latching the opposite door so that in the event that one door is accidentally left open it can be closed by remote control. The air-locks have nozzles installed which will permit pressure testing of the lock at any time.

An interior lighting system and a communications system are installed.

These systems are capable of operating from the emergency power supply.

3.8.2.1.11 Vacuum Relief

Protection of the containment vessel against excessive external pressure is provided by two independent vacuum relief lines. The arrangement of instrumentation and valving is shown on Figure 9.4-2.

Each vacuum relief assembly consists of a check valve inside and an automatic air operated butterfly valve outside the containment vessel. Actuation of the butterfly valve in controlled by differential pressure between the shield building annulus and the containment vessel. A transmitter senses the differential pressure and provides a signal to the pilot solenoid on the air operated butterfly valve to open the valve at a differential pressure of 2.5 in. wg. The other switch provides an alarm signal at 4 in. wg. The check valve is counter weight balanced to open at a differential pressure of 1.1 in. wg.

The Regulatory staff has requested a series of parametric studies for conditions more severe than above. Section 6.2.1.1 demonstrates the acceptability of the valves.

3.8.2.1.12 Containment Vessel Coating

The original coating system used on the containment vessel is as follows:

a) A prime coat of inorganic zinc paint of an average dry film thickness of 2.5 mils.

b) A finish coat of phenolic paint of 4 mils dry film thickness.

Before selecting this coating system sample coupons of SA-516 Gr 70 material of the same nominal thickness and with the same surface preparation as the containment vessel shell were coated with an average dry film thickness of 2.5 mils of inorganic zinc paint. These coupons were placed in a pressure vessel about to be post weld heat treated and subjected to the same heat and for the same holding time as was the containment vessel, using a similar type of heat source.

When cool, the coupons were examined for deterioration of the coating, and there was no evidence of this. The coupons were then coated with 4 mils of phenolic paint and subjected to simulated LOCA conditions. Test reports show that the LOCA conditions had no degrading effect on the coating system either by flaking, delamination, peeling, blistering or chalking. Refer to Section 3.8.3.6.1 for more information on coatings used inside the containment.

3.8.2.1.13 Structural Pre-Operational Quality Control and Testing

a) General Requirements

Test, code and cleanliness requirements accompany each specification or purchase order for materials and equipment. Tests to be performed by the manufacturers are enumerated in the specifications together with, the requirements, if any, for test witnessing by inspectors. Fabrication and cleanliness standards, including final cleaning and sealing are also described together with shipping procedures. Standards and tests are specified in accordance with applicable regulations, recognized technical society codes and current industrial practices.

The containment vessel manufacturer is required to submit design calculations, drawings, and weld procedures to the applicant for review by the engineer/constructor before the performance of any work. This review, and review of work during construction, assures compliance with applicable codes and specifications.

All welders and welding procedures are qualified in strict accordance with, and meet the requirements of Section IX of the ASME Code. Prior to the start of welding operations, the vessel manufacturer provided the applicant and his engineer with copies of the qualified welding procedure specifications and reports of the results of the qualification tests for each welder or welding operator.

All longitudinal and circumferential welds in the shell of the containment vessel are double-welded full penetration butt joints. All butt joints in any accessories subject to the ASME Code are full penetration welds.

All welds subject to the ASME Code are 100 percent radiographed or otherwise examined in accordance with the ASME Code. Welds which cannot be radiographed, or where the interpretation of radiographs would be open to doubt are examined by the magnetic particle, liquid penetrant or ultrasonic method.

In manual arc-welding, the electrodes are of the low hydrogen type. All welding filler metal has mechanical properties which are similar to the base metal. All automatic welding is by the submerged-arc process.

Preheat in accordance with the ASME Code is applied to all seams whose thickness exceeds 1-1/4 inch regardless of the surrounding air temperature. Preheat at 100 F is applied to thinner seams if the surrounding air temperature falls below 40 F and/or the surfaces to be welded are damp.

b) Post-Weld Heat Treatment

Post-weld heat treatment was performed as required by and in accordance with the ASME Code. For field post-weld heat treatment, after the vessel was completely erected and welded, it was externally insulated with a temporary blanket type insulation suitable for the post-weld heat treatment operation, and attached mainly by banding. Temporary supports, covers and insulation, required for effecting the post-weld heat treatment operation were attached with a minimum of welding to the vessel.

Thermocouples of the Iron-Constantan type were used to monitor temperatures during the post-weld heat treatment operation, and were located as to indicate representative temperatures of areas of the vessel. Thermocouples were used to monitor temperatures of the vessel shell and bottom head, and to serve as control points on the top head (which does not require heat treatment because of lesser thickness) during the heat treatment cycle. The thermocouples were attached by welding to the outside surfaces of the vessel. The hot or measuring junctions of the thermocouples were protected by special sleeves welded to the part being monitored.

The heating of the vessel was done with luminous flame oil burners firing through openings in the bottom and sides of the vessel and arranged in such a way that the heat was evenly distributed throughout the vessel during the heat-up and holding periods without flame impingement on any part of the vessel. Combustion products were exhausted through an opening in the top of the vessel.

Temperatures at the thermocouple locations were simultaneously recorded against time on a direct reading strip chart using multiple point potentiometer type instruments.

Heat-up rates, holding temperatures and times, cool-down rates and temperature gradient restrictions were in accordance with Section III, Subsection B, of the ASME Pressure Vessel Code.

During heat-up from ambient to holding temperature (1150F), the vessel became approximately 14 inches larger in diameter. Special attention was given the temporary peripheral supporting columns during the post-weld heat treatment cycle. In order to prevent the development of excessive stresses at the columns-to-vessel connections and in the columns themselves, provisions were made for the bases of the columns to move radially outward during the heat-up period and inward during the cool-down.

Upon completion of the post-weld heat treatment operation, the insulation and other temporary items were removed and temporary attachment weldments ground smooth.

c) Materials Testing

Charpy V-Notch impact tests are made on material, weld deposit, and the base metal weld heat affected zone employing a test temperature of not higher than 0 F. The requirements of the ASME Code, Paragraph N-1211 are met for all materials under jurisdiction of the code.

Impact test of weld deposit and base metal weld heat affected zone is made for each welding procedure requiring ASME Code, Section IX qualifications.

Specimen removal from the test weld must conform to the requirements of ASME Code Section IX and removal of the impact specimens is in accordance with Paragraph N-541.3.

d) Pressure Testing

On completion of fabrication and post-weld heat treatment of the containment vessel, and prior to the installation of penetration internals, pneumatic tests are performed in accordance with the applicable requirements of the ASME Code to demonstrate the integrity and leak tightness of the completed vessel. All testing is performed prior to the concrete fill-being placed under the vessel.

A soap bubble inspection test is conducted with the vessel pressurized to 5 psig. Soap suds are applied to all weld seams and gaskets, including both doors of the personnel air locks. A second soap bubble inspection test is performed at 39.6 psig upon completion of the over-pressure test in accordance with the requirements of the ASME Code. After successful completion of the initial soap bubble test, a pneumatic pressure test is made on the containment vessel and each of the personnel air locks at a pressure of 49.5 psig. Both the inner and outer doors of the personnel air locks are tested at this pressure. The test pressure in the containment vessel is maintained for at least one hour. The test pressure is maintained on each individual airlock door for at least one-half hour. Following a successful completion of the over-pressure test, a leakage test at 44 psig pressure is performed on the containment vessel with the personnel airlock inner doors closed. Pressure is maintained for whatever length of time is required to demonstrate full compliance with the leak tightness requirements. The leakage rate is determined by the "reference system method" which consists of measuring the pressure differential between the contained air and that of a hermetically closed reference system within the containment vessel.

The equipment used is capable of measuring with an accuracy consistent with the measurements to be made. Continuous hourly readings are taken until it is satisfactorily shown that the total leakage during any 24-hour period does not exceed 0.2 percent of the total contained weight of air.

The tests of the airlocks include operational testing and an overpressure test.

After completion of the airlocks, including all latching mechanisms and interlocks, each airlock is given an operational test consisting of repeated operation of each door and mechanism to determine that all parts are operating smoothly without binding or other defects. All defects encountered are corrected and retested. The process of testing, correcting defects, and retesting is continued until no defects are detectable.

The airlocks are pressurized with air to 49.5 psig. All welds and seals are observed for visual signs of distress or noticeable leakage. The airlock pressure is then reduced to 39.6 psig, and a soap solution is applied to all welds and seals and observed for bubbles or dry flaking as indications of leaks. All leaks and questionable areas are clearly marked for identification and subsequent repair.

The internal pressure of the airlock is reduced to atmospheric pressure and all leaks repaired after which the airlock is again pressurized to 39.6 psig with air and all areas suspected or known to have leaked during the previous test are retested by above soap bubble technique. This procedure is repeated until no leaks are discernible by this means of testing.

e) Penetrations

Penetration closure devices for electrical and piping penetrations are purchased by written specification from suppliers with tested closure devices for similar service. Performance data from prototype closures of similar or identical design is required as part of vendor qualifications.

Pipe penetrations which must accommodate thermal movement are provided with expansion bellows. The bellows expansion joints are designed to withstand containment vessel maximum internal pressure and can be checked for leak-tightness when the containment vessel is pressurized. In addition, these joints are provided with a second seal and test tap so that the space between the seals can be pressurized to the maximum internal pressure to permit testing the individual penetrations for leakage at any time.

Penetrations which are welded directly to the containment vessel can be leak tested by pressurizing the entire containment vessel.

Electrical penetrations are also provided with double seals and are separately tested. The test taps and seals are so located that the leakage tests of the electrical penetrations can be conducted without entering or pressurizing the containment vessel. Electrical penetration assemblies are tested in accordance with IEEE-317.

All containment closures which are fitted with resilient seals or gaskets are separately tested to verify leak tightness. The covers on flanged closures are provided with double seals and with a test tap which will allow pressurizing the space between the seals without pressurizing the entire containment system. In addition, provision is made so that the space between the airlock doors can be pressurized to full containment vessel maximum internal pressure.

3.8.2.1.14 Post-Operational Testing and Inspection

a) Leakage Rate Testing

Periodic leakage rate tests of the containment vessel and leak tests of the testable penetrations will be conducted as described in Section 6.2 to verify their continued leak-tight integrity.

b) Surveillance of Structural Integrity

A steel shell pressure containment vessel, designed, fabricated, inspected and pressure tested in accordance with the ASME Boiler and Pressure Vessel Code and protected by the-concrete shield building will offer continued structural integrity over the life of the unit. The vessel receives a code stamp from an authoritative body and represents the most recent developments in the techniques of pressure vessel design and fabrication that are backed up by years of research, testing and successful in-service experience. Therefore, it is contemplated that there will be no need for any special in-service surveillance program other than visual inspection of the exposed interior and exterior surfaces of the containment vessel.

3.8.2.2 Shield Building

3.8.2.2.1 Description

The shield building is a reinforced concrete structure of right cylinder configuration with a shallow dome roof surrounding the containment vessel. An annular space of 4 ft minimum is provided between the containment vessel and the interior face of the concrete shield building to permit construction operations and periodic visual inspection of the steel containment vessel. The volume contained within this annulus is 543,000 cu. ft.

The shield building has a height of 230.5 feet measured from the top of foundation base to the top of the dome. The structure consists of cylinder wall measuring 200 feet from the base to the springline of the dome with an inside diameter of 148 feet. The cylinder wall is 3' 0 thick except for two different thickness, i.e., 5 feet and 8 feet, in the area of 15 feet from the bottom of the wall. The dome consists of 2.5 feet thick concrete with an inside radius of 112 feet.

3.8.2.2.2 Design Conditions

The design of the shield building provides for biological shielding, controlled release of the annulus atmosphere under accident conditions, and environmental protection of the containment vessel.

The following loadings are considered in the design of the shield building:

a) Dead Loads

Dead load consists of the dead weight of the shield building, the containment vessel, grout under the containment vessel, the foundation slab, the weight of structural steel and the concrete interior structure within the containment vessel.

Densities used for dead load calculation are as follows:

1) Concrete: 138 lb/cu ft

2) Steel reinforcing: 489 lb/cu ft using nominal cross-section areas of reinforcing bar sizes.

3) Steel containment vessel: 489 lb/cu ft

b) Loss of Coolant Accident Load

The loss of coolant accident load is determined by analysis of the pressure and temperature transients in the annulus during a loss of coolant accident (LOCA). The shield building ventilation system is designed to keep pressure in the annulus within design pressure under this condition (see Section 6.2).

c) Live Load

Live load on the dome which is uniformly applied to the top surface of dome at an assumed value of 30 lb per horizontal plan projection square foot in accordance with the local building code. Live load is not considered in conjunction with loss of coolant accident load, and is intended only to assure structural adequacy of the roof.

d) Hurricane Wind Load

Hurricane wind loading for the shield building is based on 194 mph wind including gust factors with no variation of wind velocity with height, acting concurrently with an additional 1.5 psi differential pressure.

e) Tornado Load

The shield building has been analyzed for tornado loading (not coincident with earthquake) on the following basis to insure no loss of function.

Lateral force on the shield building is assumed as the force caused by a tornado having a 300 mph peripheral tangential velocity plus a 60 mph translational velocity for a total of 360 mph, together with a differential pressure of 3.0 psi between the inside and the outside of the shield building applied as a static pressure (see Section 3.3.2).

f) Temperature Load

The shield building does not feel the thermal effects of the accident. But the normal operating thermal gradients across the wall and dome will cause thermal expansion stresses. Two extreme conditions were considered.

1) Summer operation:

    a) Operating temperature inside containment = 120 F

    b) Exterior sustained concrete temperature = 95 F

2) Winter operation:

    a) Operating temperature inside containment = 95 F

    b) Exterior sustained concrete temperature = 35 F

In any case, the "as constructed" temperature was assumed at 70 F.

g) Uplift Due to Buoyant Forces

Uplift forces which are created by the displacement of ground water by the structure have been accounted for in the design.

h) Seismic Loads

Seismic loads are computed using the following:

1) OBE horizontal seismic ground acceleration is 0.05 g.

2) DBE seismic ground acceleration is 0.10 g.

3) Vertical component of 2/3 of the horizontal ground acceleration is applied simultaneously with the horizontal acceleration.

The seismic analysis has been based on a structural dynamic analysis as discussed in Section 3.7.2.

i) External Missiles

The shield building is designed to withstand, without loss of function, a tornado driven missile equivalent to a 10 foot long 2 inch x 4 inch timber traveling at 360 mph, or a 4000 lb automobile traveling at 50 mph. Turbine missiles have been considered (see Section 3.5).

3.8.2.2.3 Leakage Rate

The shield building is designed so that its leakage rate at 1/4 inch of water is not greater than the quantities indicated in Table 6.2-2.

3.8.2.2.4 Codes

The shield building is designed in accordance with applicable portions of the ACI Code 318-63 and state and local building code requirements.

3.8.2.2.5 Materials

Specifications and working drawings for materials and their installation are of such scope and detail as to assure the desired integrity of the shield building.

a) Concrete

The aggregate as a minimum conforms to all the requirements of "Specification for Concrete Aggregates" ASTM-C33. Mixing water is required to pass ACI 318 requirements.

Concrete tests mixes are determined using the same materials being furnished for the job. Test cylinders are prepared and tested with results studied when 7 and 28 days old.

Air entraining agents are used. Retarding agents are added when field construction conditions make it desirable. Air entraining admixtures conform to ASTM-C 260. Retarding agents conform to ASTM-C 494. In no case is calcium chloride used in admixture.

b) Reinforcement

All required reinforcing is new billet steel in accordance with ASTM A615-68.

Mill tests results are obtained from the reinforcing steel supplier for each heat of steel to show proof that the reinforcing steel has the specified composition, strength and ductility. Bars No. 11 and smaller are lap spliced in accordance with ACI 318-63, Section 805.

No cadweld splices are used.

All reinforcing steel is shipped to the job in bundles bearing a tag identifying its size, grade and code number keyed to heat numbers. This information is verified by certified mill test reports which accompany each shipment of reinforcing steel.

3.8.2.2.6 Load Combinations

The design is based upon limiting load factors which are used as the ratio by which loads are multiplied for design purposes to ensure that the load/deformation behavior of the structure is one of elastic, small strain behavior at the design load. The load factor approach was used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. In the case of the shield building, therefore, this approach places minimum emphasis on the fixed gravity loads and maximum emphasis on earthquake or wind loads. The loads hereafter referred to as factored loads, utilized to determine the required limiting capacity of any structural element of the shield building were computed as follows:

a) $(1.0 \pm 0.05) (D + T) + 1.25 LOCA + 1.25 OBE$

b) $(1.0 \pm 0.05) (D + T) + 1.25 LOCA + 1.25 Hu$

c) $(1.0 \pm 0.05) (D + T) + 1.25 OBE$

d) $(1.0 \pm 0.05) (D + T) + 1.25 Hu$

e) $(1.0 \pm 0.05) (D + T) + 1.0 LOCA + 1.0 DBE$

f) $(1.0 \pm 0.05) (D + T) + 1.0 DBE$

g) $(1.0 \pm 0.05) (D + T) + 1.0 W$

h) $(1.0 \pm 0.05) (D + T) + 1.5 LOCA$

i) $1.5 (D + T) + 1.8 L$

j) $1.25 (D + T) + 1.25 P$

Where:

D = Dead Loads of structures and equipment plus any other permanent loading contributing stress, such as hydrostatic or soil.

L = Live load.

OBE = Operating basis earthquake load.

DBE = Design basis earthquake load.

W = Tornado load including differential pressure.

P = Internal pressure not associated with tornado.

Hu = Hurricane load including differential pressure.

LOCA = Loss of coolant accident loads including pressure and temperature loads acting on the shield building.

T = Thermal loads due to normal operating temperature gradient through the walls.

3.8.2.2.7 Load Factors

The following are descriptions for load factors used in the design equations in Section 3.8.2.2.6:

a) Dead Load

The dead load factor for the shield building design is 1.0 ± 0.05 in combination with the other factored loads. The reason for using this load factor is that the dead weight of the shield building can be accurately determined because of its simple geometric configuration. The deviations of the dead load due to construction tolerances and the uncertainties of attached equipment and piping loads is within 5 percent of the dead weight. Nonetheless, the ACI loading recommendations are also checked (load cases (i) and (j) in Section 3.8.2.2.6).

b) Live Load

In general live load is not present during plant operation. Live load is considered only during shutdown. A factor of 1.8 is used for the live load in accordance with the ACI 318-63 code in investigating this condition.

c) Earthquake Loads

For the OBE of 0.05 g ground acceleration, a factor of 1.25 is used in combination with the other factored loads in designing both the shield building and the containment internal structure. The selection of this load factor is in agreement with past and current practice of concrete containment design for nuclear power plants and also the ACI 318-63 code. Under the DBE condition of 0.10 g ground acceleration a factor of 1.0 is used both for the shield building and the containment internal structure.

The factor of unity is consistent with the loading condition which is to demonstrate no loss of function under a maximum potential loading condition.

d) Hurricane Load

The shield building is designed to withstand the hurricane load and the associated negative pressure. The design load factor is 1.25 in combination with the other factored loads in accordance with the ACI 318-63 code.

e) Tornado Load

The shield building is designed to withstand the tornado load and the associated pressure differential without loss of function. A load factor of 1.0 is used and is consistent with the loading condition and demonstrates no loss of function under tornado loading.

f) Temperature Load

The shield building and the containment internal structure are designed for thermal loads (temperature gradients) in combination with the other factored loads. Accurate extremes can be determined in establishing the temperature gradient through walls, dome and slabs. Therefore a thermal load factor of unity with variations of ± 5 percent is used for the shield building. The variations in load factors are consistent with the degree of structural complexities in the structures, and are considered in the same category as dead loads.

g) Loss of Coolant Accident Load

The steel containment vessel practically isolates the shield building from the reactor coolant system and therefore eliminates significant pressure and temperature loads on the shield building during an accident. However, small pressure build-ups and temperature changes do occur in the annulus during the accident. These are taken into account in design. The load factor for the LOCA is 1.25 in combination with the other factored loads. The shield building is also analyzed for one and one-half times the LOCA load for no loss of function.

h) Miscellaneous Loads

During a loss of coolant accident, the containment internal structure is subjected to such accident loads as pressure, thermal, internal missile, jet force, and piping anchor loads. Load factors are assigned to different load sources in loading combinations as shown in Section 3.8.2.2.6. The load factors are assigned recognizing the degree of accuracy available in determining the loading and also the unlikely combination of simultaneous load occurrences.

3.8.2.2.8 Allowable Stress Criteria

Using the factored load combinations as defined in Section 3.8.2.2.6, the various components of the shield building structure have the required load capacity when the stresses in them do not exceed the yield strengths of the materials used. The yield capacity of all load carrying structural elements has been reduced by a yield capacity reduction factor (φ) as given below.

Yield Capacity Reduction Factors:

a) φ = 0.90 for concrete in flexure.

b) φ = 0.85 for diagonal tension, bond and anchorage concrete.

c) φ = 0.75 for spirally reinforced concrete compression members.

d) φ = 0.70 for tied compression members.

e) φ = 0.90 for fabricated structural steel.

f) φ = 0.90 for mild reinforcing steel in tension (excluding splices).

g) φ = 0.90 for mild reinforcing steel in tension with mechanical splices.

The φ factors are provided to allow for variations in materials and workmanship. In the ACI Code 318-63, (p varies with the type of stress or member considered; that is, with flexure, bond or shear stress, or compression.

The φ factor is multiplied into the basic strength equation or, for shear, into the basic permissible unit shear to obtain the dependable strength. The basic strength equation gives the "ideal" strength assuming materials are as strong as specified, sizes are as shown on the drawings, the workmanship is excellent, and the strength equation itself is theoretically correct. The practical, dependable strength may be something less since all these factors vary.

The ACI Code provides for these variables by using the first four factors listed.

The additional 0 values used represent the engineer constructor's best judgment of how much understrength should be assigned to each material and condition not covered directly by the ACI Code. The additional φ factors have been selected based on material quality in relation to the existing φ factors.

Coventional concrete design of beams requires that the design be controlled by yielding of the tensile reinforcing steel. This steel is generally spliced by lapping in an area of reduced tension. For members in flexure, ACI uses φ = 0.90. The same reasoning has been applied in assigning a value of φ = 0.90 to reinforcing steel in tension, which now includes axial tension. However, the code recognizes the possibility of reduced bond of bars at the laps by specifying a φ of 0.85. Mechanical splices will develop at least 125 percent of the yield strength of the reinforcing steel. Therefore, φ = 0.90 is recommended for this type of splice.

3.8.2.2.9 Design Analysis Methods

a) Structural Model

The SHELLS program is used to analyze the shield building design. The structural model used in the analysis is shown in Figure 3.8-14. The model consists of four regions to adequately represent the geometrical configuration of the shell. The SHELLS program utilizes a finite difference approach for the solution of the reduced shell equations. Each region is divided into equal finite increments referred to as stations.

In this model Region I represents the dome with 85 stations at 11.1 inch increments. An eccentricity of 0.13 inches is introduced at the end of the region because of the centerline offset between the dome and the shell.

Region II extends from the dome (elevation 2412 inches) to elevation 1356 inches and has 89 stations (12 inch increments each). This region ends far from the openings in the lower part of the shell so that the holes will have negligible effect on the internal loads.

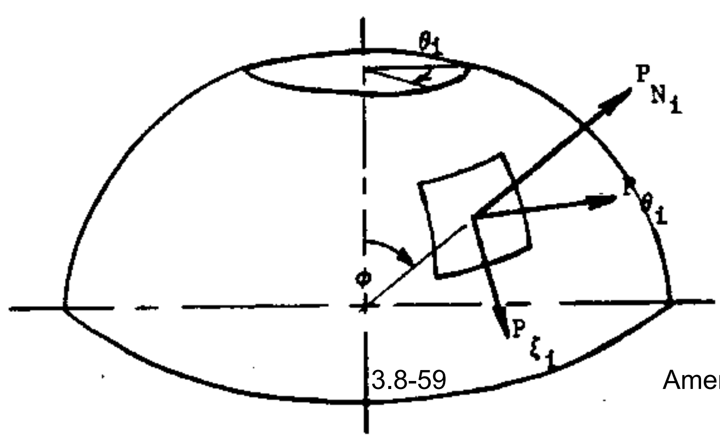
Region III extends from Region II to the start of the region of abrupt change in wall thickness at elevation 186 inches. This region has 99 stations of 11.93 inch increments each.

Region IV represents the thick portion of the shell (60 inch wall thickness) and extends from Region III (186 inches) to the elevation ‘0' (slab interface). This region is divided into 17 stations of 11.625 inch increments each.

The boundary conditions used were: a closed apex at the start of Region 1, and completely clamped ends at the end of Region IV approximating a rigid connection to the bottom slab.

b) Loading Conditions

The SHELLS program accepts external loads in the form of applied pressures. Nine loading conditions were analyzed; all loads were converted into pressure loads in psi units. The nomenclature used to describe the input of these pressures is described in the following figure:



The terms used for the i[th] station are defined as follows:

P_{N_i} = Normal pressure

P_{θ_i} = Circumferential pressure

P_{ξ_i} = Meridional pressure

Each component of pressure is shown in its assumed positive direction.

1) Condition 1, Dead Load.

A uniform dead load of 0.15 lb/ft^3 (0.0867 lb/in^3) was applied to the entire shell model as shown in Figure 3.8-15. The pressure distribution for each region is:

Region Region

Region

I: P_{N_i} = -2.6 cos φ_i

II & III: P_{N_i} = 0 IV:

P_{N_i} = 0

P_{θ_i} = 0

P_{θ_i} = 0 P_{θ_i} = 0

P_{ξ_i} = 2.6 sin φ_i

P_{ξ_i} = 3.12 P_{ξ_i} = 5.21

2) Condition 2, Live Load

A uniform live load of 0.03 ksf (0.208 psi) was applied vertically to the dome (Region I) only, as shown in Figure 3.8-16. The pressure distribution for each region is:

Region I: P_{N_i} = -.208 cos φ_i Regions II, III and IV are all zero.

P_{θ_i} = 0

P_{ξ_i} = .208 sin φ_i

3) Condition 3, Uniform Internal Pressure Load

A uniform internal pressure of 0.432 ksf (3.0 psi) was applied to the structure on Regions I, II, and Region III as shown in Figure 3.8-17; the load was applied only above grade level, station 91.

4) Conditions 4 and 5, Tornado and Hurricane Loads

Two independent wind loads were applied to the structure: a tornado load at 300 mph, and a hurricane load at 194 mph. Both loads have the same distribution over the structure, i.e. symmetric about one centerline as shown in Figure 3.8-18. The pressure distribution is based on four terms of Fourier expansion as follows:

$$P = P_0 + P_1 \cos \theta + P_2 \cos 2\theta + P_3 \cos 3\theta$$

The values of these coefficients for each region for the tornado case are:

Region I: $P_0 = -2.02$ ($P_1 = P_2 = P_3 = 0$)

Regions II and III (to station 91): $P_0 = 1.277$

$P_1 = -.586$

$P_2 = -1.947$

$P_3 = -.826$

For the hurricane case the coefficients are:

Region I: $P_0 = -.674$ ($P_1 = P_2 = P_3 = 0$)

Regions II and III (to station 91): $P_0 = .4225$

$P_1 = -.195$

$P_2 = -.6461$

$P_3 = -.2747$

No loads were applied below ground level.

5) Condition 6, Temperature Loads

Two temperature cases were analyzed as shown in Figure 3.8-19. Case I considers a unit temperature gradient. It was assumed the temperature increased inward. The reference temperature for the external environment was taken as zero and the internal environment as 1 F (i.e. providing the required unit gradient). Case 2 considers a unit homogeneous increase of temperature. This is accomplished by assuming a zero reference temperature and a 1 F temperature uniformly distributed throughout all four regions.

6) Conditions 7 and 8, Lateral Loads Due to Soils and Ground Water

Two cases of soil and ground water pressure were considered as shown in Figure 3.8-20. These loads were assumed to be uniformly distributed in the circumferential direction of the shell. For each case the water and ground pressures were superimposed to create a single load per case as shown in details in Figure 3.8-21. Note that elevation '0' is the interface of the shell with the bottom slab.

7) Condition 9, Seismic Load

The seismic load was applied to the structure as shown in Figure 3.8-22. The pressure distribution in the circumferential direction is:

$$p = -P_1\cos\theta + P_1\sin\theta$$

The values of P_1 for each are a function of G_i, where G_i is the acceleration at station i.

Region I: $P_1 = 2.6\ G_i$

Regions II and III: $P_1 = 3.13\ G_i$

Region IV: $P_1 = 5.2\ G_i$ (G_i= .136)

The seismic analysis of the shield building is discussed in detail in Section 3.7.2.

The results of the structural analysis are discussed in Section 3.8.2.2.11.

3.8.2.2.10 Structural Design

The reinforced concrete shield building and foundation mat are designed by the Ultimate Strength Design Method in accordance with ACI 318-63. The summation of loads with load factors as shown in Section 3.8.2.2.6 provides the design basis of the shield building.

Steel reinforcing is placed in the concrete walls, dome and foundation to control cracking due to concrete shrinkage and temperature gradients.

The design of the shield building is based on an elastic behavior of steel reinforcement during design basis earthquake controlling cracking of concrete and impairment of leak tight integrity.

The personnel and equipment hatch openings and the major piping penetrations through the shield building are designed such that all the anticipated loads are carried by frame action around the openings. This frame action is achieved by adding sufficient reinforcement around the perimeter of the openings. Diagonal bars at each corner of the opening are added to provide the horizontal and vertical shear resistance.

a) Foundation Slab

The foundation slab as shown on Figure 3.8-23 consists of a 10 feet thick dish shaped concrete slab with an over-all diameter of 160 feet. A circular flat slab of 57 feet 6 inches diameter is located in the center portion of the slab inclining up with a slope 4 horizontal to 1 vertical in an elevation difference of 10.5 feet.

The foundation slab of the containment vessel and shield building has no structural continuity with the foundation slab of any adjacent building.

The design provides for the relative static settlements that could occur between structures as dead load is placed on the slab during construction.

The design considers the elastic relative settlement between structures during earthquake and the relative horizontal deformations between structures due to the rocking and translation motions of the foundations as well as the elastic properties of structures.

The concrete in the foundation slab has 4,000 psi compressive strength. All reinforcing steel is ASTM A615-68 (40,000 psi yield strength). Design of the foundation slab is in accordance with applicable portions of ACI 318-63 Part IV-B.

Stress analysis of the foundation slab was performed using the finite element method. A loading combination of dead load, tornado and summer temperature gradient is governing for design in the most portions of the slab.

Reinforcing steel, as shown in Figure 3.8-24 is placed in radial and circumferential directions except that a two-way reinforced circular core of 14 feet diameter is placed in the center portion of the slab. The radial and hoop reinforcing steel is so designed to resist the radial loads and hoop loads respectivity.

During the construction stage, the cylinder wall is placed up to the spring line of the dome before the dewatering process is terminated to overcome buoyancy forces. The stresses on the slab have been investigated for this construction stage. However, this initial design was found not to govern the design of the slab.

b) Cylinder Wall

The cylinder wall is 148 feet inside diameter, 200 feet from the top of foundation base to the springline of the dome. The wall is 3 feet-0 inches thick except in the area of 15 feet on the bottom of the wall where there are two different thicknesses, i.e., 5 feet and 8 feet. Figures 3.8-25 and 3.8-26 show the masonry dimensions and outline.

Both the cylinder wall and dome are designed to resist the membrane stresses imposed by the combination condition of dead load, temperature and tornado force, and local shears and moments resulting from secondary loads caused by discontinuities.

The hoop and vertical reinforcing steel of the cylinder wall is ASTM A-615-68 new billet steel. The concrete in the cylinder wall is 4,000 psi compressive strength.

The analytical method being applied to the cylinder wall and dome for axisymmetric and non-axisymmetric loads is the SHELLS program which uses a finite difference approach for the solution of the reduced shell equations. The structural model and input loading conditions used in the analysis are given in Section 3.8.2.2.9.

The results of the structural analysis are discussed in Section 3.8.2.2.11.

The shield building is designed within the elastic limit of the building materials for all loading combinations. The applicable sections of the ultimate strength design method of ACI 318-63 and ACI Title No. 61-59, "Concrete Shell Structures Practice and Commentary," have been used for the design of the shell structure.

The reinforcing steel in the cylinder wall has been distributed in a vertical hoop pattern. The vertical and hoop reinforcing steels are designed to resist the vertical and tangential stresses respectively. In addition, a principle stress analysis has been pursued in areas of high in-plane shear. The results of this analysis were used to verify the adequacy of the reinforcing layout as acknowledged by ACI Committee 334 in the Report Title No. 61-59 when the reinforcing direction having a deviation from the direction of the principle stress is more than 15 degrees. The allowable stress decrease of 5 percent is considered to compensate for each additional degree of the deviation above 15 degrees.

The provisions for reinforcement lap splicing and stagger have been in accordance with ACI Code 318-63 para. 805 (b) for deformed bars having a specified yield strength of 40,000 psi and for concrete having a specified strength of 4,000 psi. Contact splices were generally used in the design where the spaced lap splices were used. The maximum spacing of bars was within one-fourth the lap length as recommended by CRSI.

The concrete in some portion of the wall is completely cracked both in vertical and horizontal direction under the most critical loading combinations. In the cracked sections, the radial and tangential shears are developed in the reinforcing steel by dowel action across the cracks (Reference 4). The shear friction across the interface of the cracks is assumed insignificant for a conservative design. The dowels are designed so as to develop a bearing stress on the concrete no greater than the ultimate bearing strength of the concrete and not to cause local crushing of the concrete at a crack. For a No. 11 steel bar, the allowable load-transfer capacity is about 3,000 lb, as recommended by ACI Committee 325 published in the July 1956 Journal of the American Concrete Institute.

For construction practice, vertical and horizontal construction joints are provided on the cylinder wall. The lifts are 10 feet high of conventional forming and divided into three vertical placements per cylinder lift at a typical 120 degrees apart with staggered vertical joints on every lift. The shear flow along the horizontal construction joint caused by the seismic and wind load is resisted by a concept of shear friction (Reference 5). Because the reinforcement is well anchored on both sides of the construction joint, any separation will develop tension in the reinforcement which provides an external clamping force on the concrete resulting in a friction force along the interface of the construction joint.

Two extreme temperature conditions for normal operation are given in Section 3.8.2.2.2. However, they were not applied in the design of the cylinder wall below grade. A constant temperature of 70 F (construction temperature) has been assumed on an el-5 feet. By linear interpolation, the average temperatures on the outer face of the 5 foot thick wall are 78.4 F in the summer and 58.4 F in the winter. The same temperature has been applied in the design of foundation base.

In the computer stress analysis, a non-cracked uniform section was assumed. The moment due to temperature gradient is:

$$M_{\Delta T} = \frac{E_c \alpha (\Delta T) h^2}{12(1-\nu)}$$

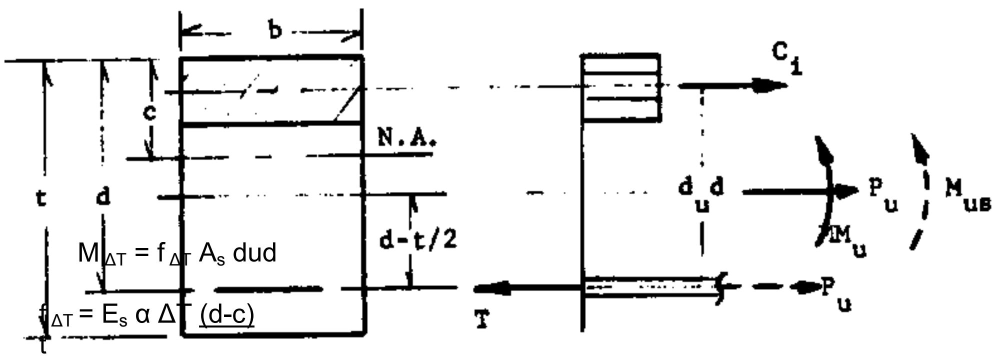
where:

E_c = modulus of elasticity of concrete (psi)
α = coefficient of linear expansion (in/F)

ΔT = difference in temperature gradient (F)
h = thickness
v = Poission's ratio

Under the most critical loading combinations, the wall is partially or completely cracked both in vertical and horizontal directions. The moments due to temperature are as follows:

1) Partially cracked section



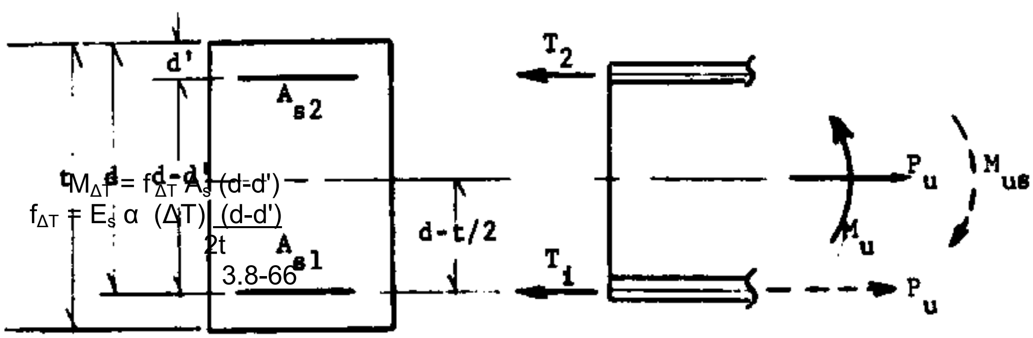
$$M_{\Delta T} = f_{\Delta T} A_s d_u d$$

$$f_{\Delta T} = E_s \alpha \Delta T \frac{(d-c)}{t}$$

where:

E_s = Modulus of Elasticity
$f_{\Delta T}$ = Steel stress
A_s = Area of reinforcing steel
b = Width of member
M_{us} = Ultimate strength moment
t = Thickness of Section

2) Completely cracked section



$$M_{\Delta T} = f_{\Delta T} A_s (d-d')$$

$$f_{\Delta T} = E_s \alpha (\Delta T) \frac{(d-d')}{2t}$$

Therefore, the temperature moment computed by computer analysis has been modified according to the partially or completely cracked section in the design of the critical loading combinations.

Each penetration of the cylinder wall creates a discontinuity around which stresses must be carried. Where the penetrations are relatively small, the reinforcing steel has been shifted slightly to clear the opening. At penetrations too large to pass reinforcing steel around by slight shifting of the bars, additional reinforcing steel has been provided to act as a frame around the perimeter of the openings. In addition, diagonal bars have been added at each corner of the openings to provide horizontal and vertical shear resistance.

The main penetrations openings in the cylinder wall are as follows:

Fuel transfer tube	5'-0 dia.
Equipment hatch	13'-0 dia.
Personnel lock	10'-9 dia.
Main steam lines	6'-0 dia.
Feedwater lines	4'-10 dia.
Construction hatch	29'-0 dia.
Bulkhead type doors	2'-3 x 7'-3
Escape lock	7'-6 dia.
Piping, ducts, electrical cable	3" to 72" dia.

The stress analysis was performed for the construction hatch, equipment hatch and personnel lock openings by Universal Analytic Inc. using the results obtained from the SHELLS program described in Section 3.8.2.2.9. All other openings smaller than 7.5 feet in diameter are treated as small openings. The stress concentration has been considered in the design by using the following simplification method:

The curvature of the cylinder wall is assumed to be neglected in the local design. According to the effect of circular holes in plates, (References 6 and 7), the stress distribution in the neighborhood of the opening will be:

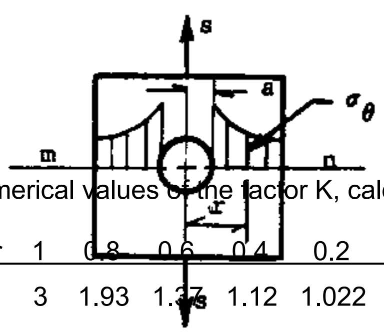
1) Under uniform tension

At section m-n

$$\sigma_\theta = \frac{s}{2}\left[2 + \frac{a^2}{r^2} + 3\frac{a^4}{r^4}\right]$$

$$\sigma_\theta = K_s$$

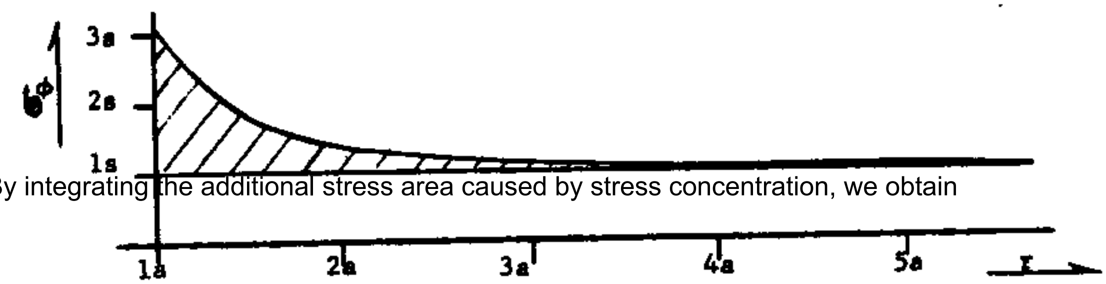
$$K = 1 + \frac{a^2}{2r^2} + 3\frac{a^4}{2r^4}$$



The numerical values of the factor K, calculated for several values of a/r, are given as follows:

a/r	1	0.8	0.6	0.4	0.2	0.1
K	3	1.93	1.37	1.12	1.022	1.01

It is evident that the effect of the opening is of a very localized character. The distribution of this stress is shown below.



By integrating the additional stress area caused by stress concentration, we obtain

$$F = \int_{o} \sigma_{\theta} dr - (10a - a)s$$

$$F = \frac{s}{2} \int_{o}^{10a} \left[2 + \frac{a^2}{r^2} + \frac{3a^4}{r^4} \right] dr - 9as = 0.95as$$

It means that 95 percent additional stress is caused by stress concentration.

2) Under bending

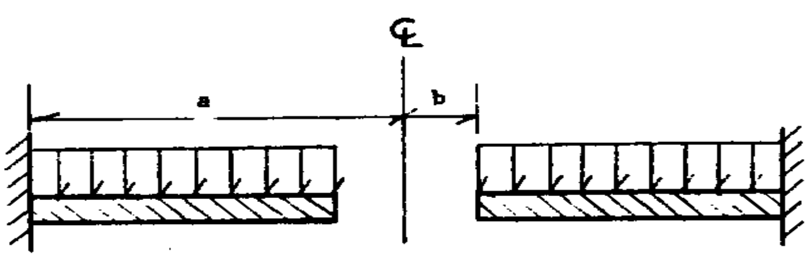
Maximum bending stress of circular plate with clamped end under uniform load w (k/ft^2):

$$(\sigma r)_{\max} = (\sigma t)_{\max} = \frac{3(1+\nu)wa^2}{8b^2} = 0.49\frac{wa^2}{b^2}$$

$(when \quad \nu = 0.3)$

Maximum bending stress of circular plate with clamped end and a circular hole at center of the plate under uniform load w (k/ft^2):

$$\sigma'_{max} = K\frac{wa^2}{b^2}$$



$For\ \nu = 0.3$, the numerical values of K are given:

a/b	1.25	1.5	2	3	4	5
K	0.105	0.259	0.480	0.657	0.710	0.730

The coefficient K is a function of (a/b) and can be expressed as follows:

$$K = A + B\frac{1}{r} + C\frac{1}{r^2} \quad (r = \frac{a}{b})$$

Using three correspondent values of K and a/b, A, B and C are:

A = 0.825
B = -0.35
C = -0.69

The K value can be rewritten as:

$$K = 0.825 - 0.35\frac{1}{r} - 0.69\frac{1}{r^2}$$

$$For\ a/b = 10 \quad K = 0.783$$

$$\sigma'_{max} = 0.783\,\frac{wa^2}{b^2}$$

The maximum bending stress is about 60 percent more than that in the plate without a center hole.

A conclusion has been made for the small opening in the cylinder wall that all reinforcing steel cut to clear the opening shall be replaced at the side of the opening such that all the anticipated loads are carried by frame action around the openings.

The cylinder wall has been analyzed to withstand, without loss of function, a tornado driven missile equivalent to a 10 ft long 2" x 4" timber traveling at 360 mph, or a 4,000 lb automobile traveling at 50 mph at a height up to 25 feet above grade.

The analysis is based upon elastic impact between the missile and the cylinder wall. Local penetration due to the high velocity missile are determined using equations suggested by Reference 8.

Due to the instantaneous nature of missile impact loading the following maximum concrete allowable stress criteria as recommended in Reference 9 were used:

Axial or flexural compression	1.25 f'_c
Shear	0.20 f'_c
Bond	0.15 f'_c

In preventing the seepage of water, 6" PVC waterstops have been used both in vertical and horizontal construction joints. In addition, the foundation base and cylinder wall up to el 17 ft (1'-0 below grade) have been completely sealed and protected by a water impermeable membrane system.

c) Dome

The dome is a spherical segment with an inside radius of 112 feet and 2'-6" thick reinforced concrete. The masonry outline of the dome is shown on Figure 3.8-27.

The dome is designed for construction conditions as well as in-service conditions. The dome was constructed in two stages. The first stage is a thin reinforced concrete dome with varying thickness 5 inches at the crown and 12 inches at the edge, supported on the steel containment vessel which has been designed to accept these construction loads. The thin dome is designed to accept the load of the remaining dome concrete without the steel containment support.

The loading conditions for the first stage are as follows:

1) Dead Load (D_1) – 150 lb/cu ft

2) Dead Load of concrete in 2nd stage (D_2)
(a) Applied uniformly, or
(b) Applied non-uniformly

3) Live Load (1)
(a) 20 psf applied uniformly, or
(b) 20 psf applied non-uniformly, or
(c) 4,000 lb concentrated load

4) Wind Load (W)
In accordance with Florida Building Code - 120 mph.

5) Hurricane Load (H) - 194 mph.

6) Shrinkage (T) - equivalent to a temperature drop of 40 F

The loading combinations and design basis for the first stage are:

1) $D_1 + D_2 + L$ (working stress design)

2) $D_1 + W + T$ (working stress design)

3) $D_1 + H$ (yield point stress design)

The design of second stage of the dome is based upon the loading combinations stated in Section 3.8.2.2.6.

Epoxy coating is applied on the construction joint between first and second stage to develop bond stress, so, the entire section has been considered in the design of second stage.

The dome is reinforced with a circular core of 14 feet in diameter at the crown of the dome stretching out in a radial - circumferential pattern as shown in Figures 3.8-28 and 3.8-29.

d) Containment Vessel Support

Concrete and grout were placed beneath the ellipsoidal bottom of the containment vessel. The general procedure is:

1) To place six feet of concrete inside the steel vessel

2) To place concrete beneath the vessel as close as possible to the bottom of the vessel

3) To pressure grout with epoxy-polysulfide between the concrete and bottom of the vessel through channels left by pulled tubing

A 4 inch thick layer of ethafoam has been placed outside the steel containment vessel at the spring line and a reinforced concrete ring girder has been designed to resist the thrust of the vessel at the spring line.

3.8.2.2.11 Calculated Results

The shield building has been analyzed for each of the loading conditions as discussed in Section 3.8.2.2.9. The resulting forces, moments, shears and deflections are shown on Figures 3.8-32 through 3.8-43.

The resulting loads for the various loading conditions are combined according to the load combination equations given in Section 3.8.2.2.6. In all cases the calculated design loads are within the ultimate capacity of the structural members. Table 3.8-11 gives a comparison of the calculated values of shear, axial load and moment for the principal structural members of the shield building and interior concrete structure. The calculated values are given for the most severe loading combination for the particular member.

3.8.2.2.12 Structural Preoperational Testing and Inspection

a) General Requirements

Appropriate ASTM Material Specifications are cited in the building specifications for all construction materials which will describe the testing and basis for acceptance of materials. Standards and tests are specified in accordance with applicable regulations and current building practices.

The testing of concrete and reinforcing bars is accomplished by an independent testing laboratory whose primary business is to perform such testing and who can show proof of the required knowledge and facilities to perform the specified tests and report accurate results.

This testing company examines local aggregates and cement, design the concrete mixes, take samples and make the required field tests.

b) Concrete Testing

Concrete for seismic Class I structures has been tested in accordance with the requirements outlined below.

The methods used in sampling, making, curing and testing the concrete samples, either in the field or in the laboratory, have been in accordance with the appropriate ASTM Standards listed below:

ASTM C172 - Method of Sampling Fresh Concrete
ASTM C31 - Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Field
ASTM C192 - Method of Making and Curing Concrete Compression and Flexure Test Specimens in the Laboratory

ASTM C39 - Method of Test for Compressive Strength of Molded Concrete Cylinders

Sampling and testing has been performed by an independent testing laboratory. All concrete samples are taken at the point of placement in accordance with ASTM C172. Initially, the number of test cylinders made were as follows:

	Min. No. of Cylinders	Test Breaks 3 Day	7 Day	28 Day	Extra
Until final determination of each design maximum	4		1	2	1
For each Concrete Placement					
10 to 50 cu. yd.	3		1	1	1
50 to 100 cu. yd.	6		2	2	2
Over 250 cu. yd.	9		3	3	3

The extra cylinders are tested only if it is necessary to substantiate 7 or 28 day test results.

After more than 150 tests were performed, the above frequency was changed to meet the requirements of ACI 318-71, Section 4.3.

The following tests have been performed for all samples:

1) Slump Test - ASTM C143

2) Air Content Test - ASTM C138

3) Temperature

4) Dry Density Test (ASTM C567)

(a) A sample is taken for each trial mix.
(b) A sample is taken for each 250 yards of concrete for Class I nuclear structures and shielding concrete.

5) Density Tests of Fresh Concrete (ASTM C138)

(a) A sample is taken for each trial mix.
(b) A sample is taken each time strength specimens are cast, one for each set of test cylinders.
(c) A sample is taken for each batch from which a sample for dry density test is taken.

6) The test for determination of yield strength (ASTM C138) is performed each time strength specimens are cast. Strength specimen samples taken in accordance with ASTM C31 and CI192 and the number of cylinders prepared is as described above. Evaluation of test results is in accordance with ACI Std. 214-65.

b) Rebar Testing

The testing reinforcing steel used in safety related concrete structures including the shield building does not conform in detail to the specific requirements given in AEC Safety Guide 15, "Testing of Reinforcing Bars for Concrete Structures," which was issued October 27, 1971. The safety guide was not in existence at the time of issuance of the plant construction permit and therefore its requirements could not be incorporated in the rebar testing program. The rebar testing performed is as follows:

Reinforcing steel samples of each bar size, at least one from each heat but not less than one per 100 tons, are supplied by the reinforcing steel supplier. These samples are tested by the user based upon ASTM specifications. The method of selecting testing samples is as follows:

1) Random sampling:

Random sampling and testing in the amount of 5 percent of the samples supplied are required. Sampling is weighted to match rebar size distribution (i.e., 5 percent of each rebar size samples used) since chemical composition varies according to size. If a heat is greater than 500 tons, at least one sample from the heat is tested.

2) Mandatory test required if mill test shows:

(a) Yield strength less than 5 percent above ASTM minimum.
(b) Ultimate strength less than 5 percent above ASTM minimum.

(c) Yield and ultimate strength radically different percentiles greater than the respective ASTM minimum.
(d) Elongation less than 10 percent greater than ASTM minimum.
(e) Weight 5 percent or more below ASTM standard.

3) All samples are tested for:

(a) Tensile yield strength
(b) Tensile ultimate strength
(c) Elongation in 8"
(d) Weight

Inspections are performed as necessary to verify compliance with specifications.

3.8.3 CONTAINMENT CONCRETE INTERNAL STRUCTURES AND SEISMIC CATEGORY I STRUCTURAL STEEL

3.8.3.1 Description

The reactor building internal structure consists of a concrete floor fill in the bottom of a steel containment vessel upon which rests a circular concrete secondary shield wall, a concrete primary shield wall, a concrete refueling cavity, and a reactor internals storage area. A concrete operating floor is supported on the secondary shield wall, the primary shield wall and part of the refueling cavity wall. The concrete floor fill, the secondary shield wall and the operating floor form a compartment within which the entire reactor coolant system is located. A general masonry outline is shown on Figures 3.8-30 and 3.8-31.

On top of the operating floor, shield walls are installed around the pressurizer and steam generators. The shield wall around each steam generator serves as a support for the upper brackets and snubbers on the steam generator. The main supports at the bottom of the steam generators and the pressurizer are on a pedestal supported directly upon the concrete fill.

On the top and bottom of the secondary shield wall, sufficient vent openings are provided to minimize the internal pressure due to a loss of coolant accident. These vent openings are protected by a concrete wall that provides radiation shielding for the opening as well as missile and jet force protection.

The refueling cavity liner is stainless steel. The function of the liner is to render the refueling cavity completely watertight against the escape of radioactive water. The refueling pool liner is welded to stainless steel embedments in the concrete floors and walls.

The reactor building internal structure is a reinforced concrete structure designed to provide the following functions:

a) Biological shield during normal operation

b) Missile shield to contain any missiles or jet forces generated within the primary coolant system and prevent their impinging upon the steel containment vessel

c) Support all equipment located within the steel containment vessel

The required minimum thickness of concrete for adequate biological shielding is specified below

a) Secondary shield wall
4.0 ft

b) Primary shield wall
6.0 ft

c) Reactor cavity wall
7.0 ft

d) Refueling cavity wall
6.0 ft

e) Operating floor
4.0 ft

f) Steam generator shield wall 2.0 ft

g) Pressurizer shield wall
2.0 ft

3.8.3.2 Applicable Codes, Standards and Specifications

Concrete design is in accordance with ACI-318-63, "Building Code Requirements for Reinforced Concrete."

The design, testing, fabrication and erection of structural steel shall be in accordance with the requirements of the American Institute of Steel Construction (AISC) specification and the AISC "Code of Standard Practice for Steel Buildings and Bridges." Structural steel material shall be ASTM A36 unless otherwise noted on drawings or specifications. Embedded liners for reactor coolant piping shall be in accordance with the American Society for Testing and Materials specifications number ASTM A-441 and ASTM A-516. The material standards used for the refueling cavity shall be as follows:

a) American Society for Testing and Materials (ASTM)

    1) ASTM A-36 Structural Steel

    2) ASTM A-240 Specification for Steel for Unfired Pressure Vessels

    3) ASTM A-572 High-Strength Low-Alloy Columbium-Vanadium Structural Steel

    4) ASTM A-992 Structural Steel Shapes

b) American Welding Society (AWS)

    1) AWS A-5.4 Corrosion-Resisting Steel Welding Rods

    2) AWS A-5.9 Corrosion-Resisting Steel Welding Rods and Bare Electrodes

c) American Society of Mechanical Engineers (ASME)

    1) Boiler and Pressure Vessel Code

The steel support structure for the cask handling crane above the north end of the fuel handling building was designed in accordance with the AISC Manual of Steel Construction Ninth Edition. The concrete foundations for the cask crane structure were designed in accordance with ACI 318-99, "Building Code Requirements for Reinforced Concrete".

3.8.3.3 Loads

3.8.3.3.1 Design Loads

The design of the reactor building internal concrete structures and seismic Category I steel structures considers the following loads both individually and in specified combinations. The loads are identified and their basic values enumerated below:

3.8.3.3.1.1 Reactor building internal concrete structures and seismic Category 1 steel structures

a) Dead Load (D)

Dead load consists of the dead weight of the concrete structure, the weight of structural steel and miscellaneous building items and permanent equipment loads.

Specific weights for dead load calculation are as follows:

1) Concrete: 138 lb/cu ft

2) Steel reinforcing: as specified in ASTM A-615

3) Structural steel:
as specified in the Steel Construction Manual of the AISC

b) Live Load (L)

Live load set on the various platforms, floors and slabs is considered in the design to ensure a structure sufficiently strong to support a random temporary load condition.

c) Equipment Load (L')

The dead weight of the various pieces of equipment, including water, steam or the other enclosed fluids, supported by the reactor building internal structure.

d) Pipe Break Accident Pressure Load (P)

The pipe break accident pressures are determined by analysis of the pressure transient inside a compartment generated by the postulated high energy pipe break for various break sizes. Refer to Section 6.2.1.

e) Normal Operating Pipe or Equipment Anchor Load (A)

The pipe or equipment anchor loads are the loads exerted upon the various structural elements by the pipe or equipment restraints for normal thermal expansion of the various piping systems during normal operation or shutdown conditions.

f) Equipment or Pipe Accident Load (Q)

These are the equivalent static loads exerted upon the various structural elements by a pipe or a piece of equipment as a result of a postulated high energy pipe break accident including an appropriate dynamic factor to account for the dynamic nature of the load. For the feedwater and main steam lines, these loads are calculated at pipe anchor points.

g) Normal Operating Thermal Load (T)

Thermal load is induced by the thermal gradient existing in a structure or across the walls between the building interior and the ambient external environment during normal operation or shutdown conditions. Both winter and summer operating conditions are considered.

h) Accident Thermal Load (T_A)

Thermal load under the thermal conditions generated by the postulated pipe break, including normal thermal load T.

i) Earthquake Loads (E,E')

Earthquake loads are computed using the following:

1) E'=loads generated by the design basis earthquake (DBE), having a horizontal seismic ground acceleration of 0.10g.
2)E=loads generated by the operating basis earthquake (OBE) having a horizontal seismic ground acceleration one half that of the DBE.

Vertical component of acceleration is 2/3 the value of the horizontal acceleration and is applied simultaneously with the horizontal acceleration.

j) Internal Missile Loads or Jet Forces (M)

Jet impingement or missile impact equivalent static loads on the structure generated by the postulated high energy pipe break, including an appropriate dynamic factor to account for the dynamic nature of the load. Missiles and missile design are described in Section 3.5.

k) Water Load (W)

The loads are exerted by the water in the refueling cavity which is filled only during reactor shutdown. The water elevation is assumed to be at el + 62 ft and the density is assumed to be 62.4 lb/cu ft.

l) Hurricane Wind Load (W_H)

The loads generated by the design hurricane wind specified for the plant. Refer to Subsection 3.3.1.

m) Tornado Wind Load (W_T)

The loads generated by the plant specific design basis tornado, including wind forces, differential pressure loads and missile impact forces where applicable. Refer to Subsection 3.3.2, Section 3.5 and Appendix 3F.

3.8.3.3.1.2 Cask Crane Support Structure

The cask handling crane steel support structure, including the runway girders, is designed and qualified for the following loads:

D = Dead load of the crane and of the support structure
L = Crane lifted load of 150 tons
I = Impact load resulting from the operation of the crane
W_O = Operating wind load resulting from a wind speed of 50 mph
W_H= Hurricane wind load resulting from a wind speed of 120 mph
W_T= Tornado wind load resulting from a wind speed of 360 mph (300 mph rotational speed plus 60 mph translational speed)
E = OBE seismic load
E' = DBE seismic load

The impact loads resulting from operation of the crane that are used in the design are enveloped by the impact load calculation methods given in the AISC Manual of Steel, Ninth Edition; CMAA Specification 70-2000; "Specifications for Top Running Bridge and Gantry Type Multiple Girder Electric Overhead Traveling Cranes", and ASME NOG-1-1998, "Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)", including May 3, 2000, addenda.

The wind loads are determined using the methods given is ASCE Paper No. 3269, "Wind Forces on Structures", 1961.

For seismic analysis, a conventional lumped mass three dimensional model is used. The mathematical model represents the coupled FHB, the crane steel support structure, and the crane, including the bridge girders, trolley, end trucks and end ties. The boundary conditions at the crane wheels are in accordance with ASME NOG-1-1998, Table NOG-4154.3-1. The OBE and DBE loads are determined in accordance with the response spectrum method of analysis given in Section 3.7.2. The horizontal seismic input motion used in the analysis is based on the response spectra described in Section 3.7.1.1. Vertical ground response spectra were developed by multiplying the horizontal direction spectra amplitudes by a factors of two-thirds. The damping values used in the analysis are consistent with Section 3.7.1.3. Strain energy proportional composite damping is conservatively calculated for structural elements having different damping based on the method described in Section 3.7.2.1.6. The modal responses are combined using the root-mean-square method specified in Section 3.7.2.1.5. The maximum earthquake motion is combined co-directionally by absolute sum with the maximum seismic loads from the vertical component of earthquake motion. The process is then repeated for the other horizontal component of earthquake motion. The results are then enveloped to obtain the overall maximum seismic loads.

3.8.3.3.2 Loading Combinations

3.8.3.3.2.1 Concrete Structures

The design is based upon limiting load factors which are used as the ratio by which loads are multiplied for design purposes to ensure that the load/deformation behavior of the structure is one of elastic, small strain behavior at the design load. The load factor approach was used in this design as a means of making a rational evaluation of the isolated factors which must be considered in assuring an adequate safety margin for the structure. This approach permits the designer to place the greatest conservatism on those loads most subject to variation and which most directly control the overall safety of the structure. In the case of internal structures therefore, this approach places minimum emphasis on the fixed gravity loads and maximum emphasis on earthquake or accident loads. The loads hereafter referred to as factored loads, utilized to determine the required limiting capacity of any structural element of the internal structures are computed as follows:

a) Normal Operating Conditions:

1) U = 1.5D + 1.5T + 1.8A

2) U = 1.25D + 1.25T + 1.25A + 1.25E

3) U = 1.0D + 1.0T + 1.0A + 1.0E'

b) Shutdown Conditions:

1) U = 1.5D + 1.5T + 1.5W + 1.8L

2) U =1.25D + 1.25T + 1.25L + 1.25W + 1.25E

3) U =1.0D + 1.0T + 1.0W + 1.0E

c) Accident Conditions:

1) U = (1.0 $\pm$ 0.10) (D + T) + 1.5P + 1.25M + 1.25Q

2) U = (1.0 $\pm$ 0.10) (D + T) + 1.25P + 1.25M + 1.25Q + 1.25E

3) U = (1.0 $\pm$ 0.10) (D + T) + 1.0P + 1.0M + 1.0Q + 1.0E'

3.8.3.3.2.2 Cask Handling Crane Concrete Support Structure

The FHB cask handling crane column foundations at the grade elevation are designed for the loads and load combinations given in Sections 3.8.1.4 and 3.8.1.5 respectively, subject to the following clarifications:

D = Dead load of the concrete structure, the crane, and the steel support structure
L = Crane lifted load of 150 tons
W_H = Hurricane wind load resulting from a wind speed of 120 mph
W_T = Tornado wind load resulting from a wind speed of 360 mph (300 mph rotational speed plus 60 mph translational speed)
I = Impact load resulting from the operation of the crane

Equipment loads (L'), operating pipe anchor loads (A), pipe accident loads (R), thermal loads (T), and external missiles (M) are not applicable to the design of the cask handling crane superstructure and column fundations. The impact load (I) is added to the normal operation combination with a factor of 1.8, as follows:

$$U = 1.5 (D + S) + 1.8 (L + I) + 1.0 B$$

The column foundations are designed in accordance with the ultimate strength design (USD) methods of ACI-318-99.

3.8.3.3.2.3 Steel Structures

Elastic working stress design methods are used in the design of the steel structures. The design of these structures is based on the following loading combinations:

a) Service Load conditions

1) D + L + L'
2) D + L + L' + E
3) D + L + L' + W_H
4) D + L + L' + T + A
5) D + L + L' + T + A + E
6) D + L + L' + T + A + W_H

b) Factored Load Conditions

1) D + L + L' + T + A + E'
2) D + L + L' + T + A + W_T
3) D + L + L' + T_A + A + P
4) D + L + L' + T_A + A + P + Q + M + E
5) D + L + L' + T_A + A + P + Q + M + E'

In the above loading combinations A, T and T_A are applied statically without consideration of a dynamic load factor.

Notes to Load Combinations:

a) Dead Load

The dead load factor for the internal structure design is 1.0 $\pm$ 0.10 in combination with the other factored loads. The reason for using this load factor is that the dead weight can be accurately determined because of its simple geometric configuration. The deviations of the dead load due to construction tolerances and the uncertainties of attached equipment and piping loads is within 5 percent of the dead weight. For normal operating conditions, a load factor of 1.5 is used, per ACI 318-63.

b) Live Load

In general live load is not present during plant operation. Live load is considered only during shutdown. A factor of 1.8 is used for the live load in accordance with the ACI 318-63 code in investigating this condition.

c) Earthquake Loads

For the OBE of 0.05 g ground acceleration, a factor of 1.25 is used in combination with the other factored loads in designing both the shield building and the containment internal structures under the DBE condition of 0.10 g ground acceleration, a factor of 1.0 is used both for the shield building and the containment internal structure.

The factor of unity is consistent with the loading condition which is to demonstrate no loss of function under a maximum potential loading condition.

d) Temperature Load

The containment internal structure is designed for thermal loads (temperature gradients) in combination with the other factored loads. Accurate extremes can be determined in establishing temperature gradient through walls and slabs. Therefore a thermal load factor of unity with variations of ∀ 10 percent is used for the internal structure. The variations in load factors are consistent with the degree of structural complexities in the structures, and are considered in the same category as dead loads.

e) Loss of Coolant Accident Load

Pressure and temperature loads are taken into account in the design. The load factor the LOCA is combined with the other factored loads considering the probability of such a combination occurring.

f) Miscellaneous Loads

During a loss of coolant accident, the containment internal structure can be subjected to such accident loads as internal missile, jet force, and piping anchor loads. Load factors are assigned to different load sources in loading combinations recognizing the degree of accuracy available in determining the loading and also the unlikely combination of simultaneous load occurrences.

3.8.3.3.2.4 Cask Handling Crane Steel Support Structure

The following load combinations and acceptance criteria are used in the design of the FHB cask handling crane steel support structure:

		Load Combination	Acceptance Criteria
a)	Normal Operation	$D + L + I + W_O$	S
b)	Design Hurricane (The crane is assumed to be in its parked position with the storage locks set)	$D + W_H$	S
c)	Operating Basis Earthquake (OBE) (Seismic-induced pendulum effects are considered in the design)	$D + L + E + W_O$	S
d)	Design Basis Earthquake (DBE) (Seismic-induced pendulum effects are considered in the design)	$D + L + E' + W_O$	The lesser of 1.6S, or 0.90 times yield stress or 0.90 times critical buckling stress
e)	Design Tornado (The crane is assumed to be in its parked position with the storage locks set)	$D + W_T$	The lesser of 1.6S, or 0.90 times yield stress or 0.90 times critical buckling stress

Acceptance criterion "S" above is the required section strength based on the elastic design methods and the normal allowable stresses defined in the AISC Manual of Steel Construction, Ninth Edition.

3.8.3.4 Design and Analysis Procedures

The stress analysis of the interior structure is performed using a finite element method using a stiffness (displacement) matrix theory. Sixteen individual loading conditions are examined. The designs are based on LOCA loading combinations.

The secondary shield wall, operating floor, refueling cavity and the shield walls for the steam generator, pressurizer and vent openings are considered monolithically as one structural unit with the boundaries assumed along the bottom of the secondary shield wall, the bottom of the refueling cavity wall and the junctures with the primary shield wall. This structural unit is analyzed by a computerized stress analysis program.

The primary shield wall is a 6 feet thick cylinder and is analyzed as a thick cylinder.

The pressurizer and steam generator supports are analyzed as short pedestals.

3.8.3.5 Structural Acceptance Criteria

3.8.3.5.1 Concrete Structures

Using the factored loads as defined in Section 3.8.3.3, the various components of the reactor building internal structure have the required load capacity if the stresses in them do not exceed the yield strengths of the materials used.

To provide for the possibility that small adverse variations in dimensions and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in a net under capacity of the structural components, the yield strengths of the individual structural members are all reduced by a reduction factor " ø " - for the design cases in accordance with Section 9.2 of ACI 318-63.

3.8.3.5.2 Steel Structures

For each of the loading combinations described in Subsection 3.8.3.3.2.2, the following defines the allowable limits which constitute the structural acceptance criteria for steel structures:

Load Combination	Limit
a1, a2, a3	S
a4, a5, a6	1.5S
b1, b2, b3, b4, b5	1.6S

S is the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC Specification.

3.8.3.6 Materials, Quality Control and Special Construction Techniques

Basically, three materials are used for the construction of the reactor building internal structure. These are concrete, reinforcing steel, and fabricated structural steel. Specifications for these materials are as follows:

a) All concrete is manufactured, placed and cured in accordance with an Ebasco standard specification

b) Reinforcing steel is new deformed billet steel testing the requirements of ASTM A-615, Grade 40. The maximum unspliced length used for design is 60 ft

c) All fabricated structural steel elements are made of ASTM A-36 steel and fabricated in accordance with the latest AISC specifications for the design, fabrication and erection of structural steel for buildings

Materials for the refueling cavity liner and embedments conform to the following specifications:

a) All stainless steel sheet and plate conform to ASTM Specification A-240, Type 304. The plate is hot-rolled, annealed and pickled. All plates of the assembled liner have as similar a character as practicable. All plates shall be smooth, free of cracks and surface imperfections.

The leak detection channels on the refueling cavity liner and embedments shall conform to ASTM Specification A-240 Type 304.

The stiffeners and other items shall conform to ASTM Specification A-36.

The weld metal to join the base metals shall be as follows:

1) AWS A-5.4 Class E 308 or AWS A-5.9 Class ER 308 to join stainless steel to stainless steel

2) AWS A-5.4 Class E 309 or AWS A-5.9 Class ER 309 to join stainless steel to carbon steel

b) Embedded liners for main coolant piping through the primary shield wall shall be of ASTM A-516, Grade 70, and other steel of ASTM A-441.

PAGE LEFT INTENTIONALLY BLANK

3.8.3.6.1 Protective Coatings Inside Containment

Coatings located within the Reactor Containment Building (RCB), which could potentially be subjected to Design Basis Accident (DBA) conditions, are referred to as Service Level I coatings.

The primary purposes of Service Level I coatings are to provide corrosion protection and a suitable surface with regard to radioactive decontamination. Since Service Level I coatings are located within the RCB, failure to adhere to the surfaces to which they are applied could hypothetically result in a larger than anticipated build-up of coating material debris at the containment sump strainers. Conceivably, such a build-up could adversely impact the flow of water through the nuclear safety-related containment sump strainers and, correspondingly, the flow of water available for the safety-related function of the Containment Spray pumps for containment cooling.

The DBA conditions to which Service level I coatings could potentially be subjected include (but are not limited to) ionizing radiation, high temperature/pressure and impingement from jets/sprays. The required worst case DBA environmental service condition inside Containment is a Loss of Coolant Accident (LOCA) for which long-term (up to one year) environmental conditions of temperature, pressure, humidity and radiation are applicable. All Service Level I coatings are laboratory tested to withstand the worst case DBA LOCA conditions in order to demonstrate that coating failure and the associated potential consequences cannot occur.

A detailed Engineering Specification documents the protective coating systems used for Service Level I applications on steel and concrete surfaces inside the RCB.

Certain parameters (i.e. thermal conductivity, thickness, and volumetric heat capacity) are used in the accident analysis of Section 6.2.1.

3.8.3.7 Testing and Inservice Surveillance Requirements

All welds are visually examined, welds exposed to water shall be further examined by the liquid penetrant method. Joint welds and welds at penetrations in plates are tested for leaks using a vacuum box and soap solution.

The complete refueling cavity liner is hydrostatically tested. Leaks, if any, are repaired and rewelded.

3.8.4 NON-CLASS 1 STRUCTURES

3.8.4.1 Turbine Building

The turbine building is a 260 feet by 130 feet structure housing the various heat exchangers, switchgear, condensate pumps, auxiliary pumps, compressors and main steam condenser. The roof of the building is an operating deck which supports a 200 ton gantry crane. The turbine-generator unit is supported on a separate concrete pedestal. The turbine building is not designed as a seismic Class I structure since it contains no equipment which is required for safe shutdown or to mitigate the consequence of an accident or whose failure would result in significant release of radioactivity.

The turbine building was reanalyzed and found to maintain its integrity for the seismic loading condition. Thus there is no potential for interaction with the Category 1 piping which runs beneath it.

The principal structures associated with the turbine building are the circulating water discharge tunnels and condenser intake and discharge blocks, and the turbine-generator foundation mat and pedestal.

The turbine building is a steel frame structure supported by a continuous soil supported reinforced concrete mat foundation. The operating deck and intermediate mezzanine levels are cast-in-place concrete slabs. The turbine building general arrangement drawings are shown on Figures 1.2-3 through 1.2-7.

The structure is designed for the following loads:

a) Mezzanine Floor

   Live Load - 150 psf

   Wind Load - as per South Florida Building Code, 40 psf min.

   Equipment Load - 50 psf

b) Operating Floor

   Live Load - 200 psf

   Wind Load - as per South Florida Building Code, 40 psf min.

   Equipment Load - 50 psf

The structural steel is designed and fabricated in accordance with AISC standards. Reinforced concrete is designed and constructed in accordance with, "ACI Standard Building Code Requirements for Reinforced Concrete," ACI 318-63, Part IV B, Ultimate Strength Design. Reinforcing steel is ASTM A615 Grade 40.

3.8.4.3 Steam Generator Blowdown Building

The steam generator blowdown (SGB) building is located east of the St. Lucie Units 1 and 2 plant island and serves both units. The SGB building is a non-seismic Class I reinforced concrete building with structural steel roofs and columns in some areas. The building occupies an approximate area of 120 feet X 80 feet with a height of 57 feet above grade except for the bay at the west end of the building which has a height of 14 feet above grade.

Three monitor (storage) tanks, each supported by a ring beam, are located outdoors at the southern end of the building. The monitor tanks occupy an approximate area of 120 feet X 60 feet.

All reinforced concrete structures are designed in accordance with ACI 318-71 requirements and those of the South Florida Building Code. Structural steel is per AISC specification.

3.8.4.4 Other Non-Class I Structures

The other miscellaneous non-Class I structures on the site include:

a) Paint and lube oil storage building

b) Gas storage building

c) Chemical storage building

d) Switchyard control building

e) Circulating water system seal well

f) Chlorination and storage building

g) Security and Records building

h) Guard stations

i) Various Maintenance buildings

j) Cask Handling Facility

These structures are designed and constructed in accordance with normal practice and applicable local building codes. The structures are located remote from the seismic Class I structures as shown on the enlarged plot plan Figure 1.2-2.

3.8.4.5 Interaction with Seismic Class I Structures

There are no structural ties between any of the seismic Class I structures and the non-Class I structures, except for the structural tie of the Cask Handling Facility upper level support steel with the Fuel Handling Building north wall. The non-Class I structures are supported on separate foundations located within the Class I compacted backfill area.

The non-Class I structures are designed in accordance with the Uniform Building Code which specifies zone zero for the plant site region.

The structural separation of the seismic Class I from non-Class I structures ensures that behavior of the non-Class I structures during a seismic disturbance will not affect the response of the Class I structures.

The Fuel Handling Building structural analysis has been reviewed to assess the impact of the structural tie with the Cask Handling Facility. This review concluded that there is no adverse impact on the Fuel Handling Building due to the structural tie with non-Class 1 Cask Handling Facility.

3.8.5 FOUNDATIONS AND CONCRETE SUPPORTS

3.8.5.1 Description of Foundations and Supports

Various concrete structures support seismic Class I equipment directly and are not part of the "base" structure. The principal functions of these concrete structures are to provide support or restraint for the equipment and to transfer loadings between the equipment and the base structure. These structures are the interface between the equipment and the base structure.

All foundations and supports carrying seismic Class I components are, however, designed as seismic Class I.

The following concrete structures are considered herein:

a) Reactor Building

1) Steam Generator Base Supports

The steam generators are supported at the bottom on a steel sliding base anchored into a rigid concrete pedestal which is keyed in the concrete base at el 18 ft of the reactor building.

Load transfer between the steel base and the surrounding concrete is as follows:

(a) Compression loads are taken by bearing on the concrete.

(b) Horizontal loads are taken by bearing in the concrete pedestal and transferred by shear in the base concrete.

(c) Uplift is transferred into the base concrete by anchor bolts connected to anchor plates. The anchor plates transfer load into the concrete by bearing, with the surrounding reinforcing steel distributing the load into the mass of base concrete.

Upper restraints at el 76 ft consist of keys to restrict Z - direction movement and horizontal rotation. Snubbers are provided to restrict X - direction movements. The restrains and snubbers are anchored into the concrete shield wall. Refer to Figure 3.8-45.

2) Pressurizer Support

The pressurizer is supported on four steel columns supported and anchored at el 29.5 ft into a concrete pedestal which is laid in a concrete base at el 18 ft of reactor building. The tops of the steel columns (el 39.92 ft) are restrained horizontally in all directions by anchoring steel plates and framewoods into the concrete shield wall. Refer to Figure 3.8-45.

Load transfer between the steel supporting structure and the surrounding concrete is as follows:

(a) Compression loads are taken in bearing; pull-out loads are taken in shear with reinforcing distributing the loads.

(b) Horizontal loads are transferred from a steel plate diaphram into embedded steel plates in the secondary shield wall and refueling cavity wall.

(c) Vertical loads are transferred from the steel columns into the base concrete; uplift loads are transferred by anchor bolts and anchor plates as mentioned above.

3) Reactor Supports

The reactor vessel is supported on a built up steel girdercolumn combination anchored into the shield concrete at el 23 ft. The built up steel column is anchored into the concrete base at el -2.92 ft of the reactor building.

Load transfer between the steel support and the surrounding concrete is as follows:

(a) Horizontal loads are transferred from the girder into the Primary Shield Wall. Load transfer occurs between the embedded portion of the steel girder and the concrete by bearing. Resulting shear is transferred into the concrete by reinforcing.

(b) Uplift loads are transferred by anchor bolts into the base concrete which is reinforced to withstand the loads. Where necessary, shear plates are used to transfer horizontal loads at the column base into the concrete by bearing.

4) Reactor Coolant Pump Supports

The reactor coolant pumps are supported at the base by compression spring supports. The pumps are laterally supported and restrained at various elevations by snubbers, wire ropes and structural steel brackets.

b) Reactor Auxiliary Building

All equipment located in the basement (el-0.5 ft) is supported by reinforced concrete piers which are tied into the base mat by a reinforced concrete pedestal. A typical foundation is shown in Figure 3.8-44. All anchor bolts are embedded into the concrete pier and pedestal.

All equipment located on structural floor slabs are constructed either on reinforced concrete pads doweled into the concrete floor slab or are welded to embedded concrete plates.

c) Fuel Handling Building

All principal equipment is supported on reinforced concrete piers which are doweled into the base slab.

d) Diesel Generator Building

The diesel generators are supported on continuous concrete pedestals doweled into the base mat.

e) Intake Structure

The cooling water pumps are supported on reinforced concrete pedestals doweled into the structure top deck.

The bridge crane support frame is supported on reinforce walls of either the intake structure or the intake canal retaining wall, or on independent footings.

f) Stream Trestles

The structural steel main steam and feedwater pipe supports are supported by reinforced concrete walls which are supported by the base mat.

The various auxiliary feedwater pumps, which are located below the trestles, are supported on individual concrete piers which rest on the base mat.

g) Component Cooling Water Heat Exchanger and Pump Area

The component cooling system heat exchangers and pumps are supported on reinforced concrete piers which rest on the base mat.

3.8.5.2 <u>Applicable Codes, Standards and Specifications</u>

All concrete is designed in accordance with ACI 318-63, "Building Code Requirements for Reinforced Concrete," ultimate strength method.

All materials used to make concrete meet the requirements of Ebasco Specification, "Concrete-Medium and Small Work." This specification is based on applicable ASTM standards. Concrete mixing and placing is in accordance with the above specification.

All structal steel is designed, fabricated and erected in accordance with AISC's "Manual of Steel Construction."

The requirements of Regulatory Guide 1.15, "Testing of Reinforcing Bars for Concrete Structures," are adhered to.

3.8.5.3 Loads and Loading Combinations

Forces resulting from the following conditions and their combinations are taken into consideration for foundation design for equipment and structures located inside buildings:

a) Normal operating loads

b) Seismic loads

c) Hypothetical seismic loads

d) Accident load conditions

The forces resulting from the following conditions and their combinations are taken into consideration for foundation design of equipment and structures located outside of buildings:

a) Normal operating loads

b) Seismic loads

c) Tornado loads

d) Wind loads

e) Hypothetical seismic loads

f) Accident load conditions

The specific load combinations used for foundations are:

1.0 (D + T)

1.0 (D + T) + 1.0 LOCA + 1.0 DBE

1.0 (D + T) + 1.0 W

A dynamic load factor of 2 is applied to the accident load condition combined.with normal operating loads and hypothetical seismic loads.

No settlements are anticipated which could effect the integrity of the structures discussed in Section 3.8.5.1.

There are no earth pressure loads involved in the structures discussed in Section 3.8.5.1.

3.8.5.4 Design and Analyses Procedures

a) Steam Generators

The foundation is analyzed for forces resulting from any one load or their combination whichever is critical. Horizontal forces are transferred by shear keys. These keys are checked for shearing and bearing stresses. Moment and uplift forces are transferred by proper reinforcement anchored into the concrete base.

All design requirements are in accordance with ACI 318-63. The design loads are:

1) X = 5648k

2) Z = 4830k

3) V = 7405k (compr)

4) U = 5828k (uplift)

All loads are inclusive of the dynamic load factor of 2.

b) Pressurizer

All horizontal loads and moments caused by horizontal loads are taken by the concrete shield walls.

The foundation is analyzed for vertical loads only. The design vertical load is 1.5 x 305.3 or 458k. Proper anchorage and reinforcement are provided to transfer loads into base concrete.

All design requirements are in accordance with ACI Code ultimate strength design.

c) Other Equipment

All parts of equipment bases which act to support the equipment are analyzed for their interaction with concrete bases. Net uplifts are taken by anchor bolts which are embedded sufficiently into the concrete bases to preclude their pullout. Where embedment is insufficient for this purpose, anchor plates are embedded. The pier reinforcing is designed to take this uplift either independently or in combination with other forces.

Horizontal shear is taken by anchor bolts where the concrete bearing strength is not exceeded. Otherwise, shear plates are incorporated into the pier-base design to transfer shear loads into bearing loads on the concrete. Shear between the concrete pier and the base mat is investigated.

Compression loads are taken directly by the piers.

Load combinations are analyzed as appropriate.

3.8.5.5 Structural Acceptance

The various concrete structures are designed within the elastic limits of the building materials for all loading conditions.

The applicable sections of the ACI 318-63 Code, as related to allowable stresses, strains, and deformations are used in the design of structures.

Differential settlement is included in the design as appropriate. Calculated differential settlements which result in significant stresses within structures are appropriately related to allowable stresses and strains in combination with the stresses and strains resulting from other loads in combination.

Shears and overturning moments are translated into stresses and strains. The design assures that the allowable stresses and strains are not exceeded.

3.8.5.6 Materials, Quality Control and Special Construction Techniques

Refer to Section 3.8.1.3.

3.8.5.7 Testing and Inservice Surveillance Requirements

Refer to Section 3.8.1.9.

3.8.6 STRUCTURAL IMPACT - HIGH ENERGY PIPE BREAKS OUTSIDE CONTAINMENT

The structural design criteria, in terms of design bases, criteria, loads, load combinations and design stress limits, is described in Section 3.8 for the reactor auxiliary building, fuel handling building, diesel generator building and intake structure. In addition, an analysis has been made for high energy pipe breaks. The criteria and other pertinent data are provided in this section and Appendices 3C and 3D. The high energy pipe break structural analysis takes into account a specific design requirement which was not previously specified in other than the more general criteria described in Section 3.8.1.

The lines which are high energy pipe lines have been analyzed for pipe break to obtain the required pressure, temperature, seismic and rupture loads, and are described in Section 3.6 and 3.9. Normal load information was obtained from pipe hanger and support details. Additionally, corrections to Section 3.8.1.5 load combinations have been made, and the balance of the requested data for fulfillment of the AEC Staff's 1973 transmittal is contained in Appendix 3D.

3.8.6.1 Structural Analysis

The results of structural analyses to support initial facility operations are complete. The structures were analyzed in accordance with design criteria described in Section 3.8.6.4. In this manner, the adequacy of the as-built structures has been established in terms of these criteria. An illustrative example of such a review is attached as Appendix 3E. The review has indicated that compliance has been demonstrated.

The required analyses for the steam generator blowdown system are presented in Appendix 3D.

The auxiliary steam system supplies 20 psig saturated steam to the boric acid and waste concentrators and various heaters within the RAB. Since the use of house heating is not likely because of the local climatology, lines for this purpose will be flanged closed before they enter the RAB as close to the header as practical (see Appendix 3D and Figure 3D-2.) The auxiliary steam lines supplying the boric acid and waste concentrators do not require restraints and the resultant blowdown from postulated ruptures will be terminated automatically with no detrimental effect on the capability of safety related structures or systems.

The letdown lines outside of containment maintain a high energy status up to and including their pressure reducing valves. Upstream conditions are 2200 psig and 450F, and downstream are 180 psig and 140F. These high energy lines (upstream portion) are seismic Class I up to the letdown control valves and are restrained as delineated in Figure 3D-1. Rupture analyses presented in Appendix 3D confirm the maintenance of structural integrity within the RAB.

The seismic Class I charging lines downstream of the charging pumps (2300 psig and 120F) are appropriately restrained. A postulated rupture of a charging line results in a rapid decay in charging system pressure. The resultant blowdown will not compromise the capability of structures. Additional information regarding this system is provided in Appendix 3D.

3.8.6.2 Load Combinations and Acceptance Criteria for Category I Steel

Protection of Category I structures, systems and components from the dynamic effects of postulated high pressure pipe breaks is achieved through the provision of pipe whip restraints at critical locations on the piping systems and/or automatic termination of blowdown.

All pipe whip restraints, with the exception of those which are part of the main steam trestle, are independent of dead and live load supports and of seismic restraints. A working stress method of design is used which satisfied the criteria set forth in Paragraph D-1 of AEC Enclosure 2.

Collar sizes for these restraints are determined from considerations of the hot and cold positions of the pipe and are large enough to permit free thermal movement of the pipe. Thus the only load a restraint experiences is Y_r, the reaction produced on the restraint by the whipping pipe. This load is increased by a factor of 2 to account for dynamic impact. Y_j is not considered, since we are assuming only one break occurs at a given time. A single restraint cannot be subjected to Y_j and Y_r together. Y_m will not occur since the location of pipe whip restraints is such as to preclude the generation of a missile by the ruptured pipe.

The main steam trestle and its restraints are subjected to normal loads as well as those due to pipe rupture. The criteria for this structure, expressed in the nomenclature of AEC enclosure 2, are as follows:

a) For those restraints which serve as thermal anchor points:

1) $S = D + L + T_o + R_o + E_{eqs}$

2) $1.5S = D + R_o + Y_r + Y_j$

3) $1.5S = L + R_o + W$ (W = Tornado Wind)

Since this is an outdoor structure, Ta, Ra and Pa are not applicable. Y_j is analyzed as a local effect only since its action on the overall structure is cancelled out by Y_r which occurs simultaneously.

3.8.6.3 Restraints

Table 3.8-13 provides a listing of the lines for which the restraints have been analyzed (historical information - applies to the time of initial plant operation). For current updated information, the reader should review line numbers and calculations in the plant's equipment database.

3.8.6.4 Design Criteria

3.8.6.4.1 Design Loads

a) NORMAL LOADS

Normal loads are those loads to be encountered during normal plant operation. They include the following:

D - Dead loads and their related moments and forces, including any permanent equipment loads.

L - Live loads, present during the pipe rupture event, and their related moments and forces.

T_o - Thermal loads during normal operating conditions.

R_o - Pipe reactions during normal operating conditions.

b) SEVERE ENVIRONMENTAL LOADS

Severe environmental loads are those loads that could infrequently be encountered during the plant life. Included in this category is:

Feqo - Loads generated by the Operating Basis Earthquake.

c) EXTREME ENVIRONMENTAL LOADS

Extreme environmental loads are those loads which are credible but are highly improbable. They include:

Feqs - Loads generated by the Safe Shutdown Earthquake.

d) ABNORMAL LOADS

Abnormal loads are those loads generated by a postulated high-energy pipe break accident within a building and/or compartment thereof. Included in this category are the following:

P_a - Pressure equivalent static load within or across a compartment and/or building, generated by a postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.

T_a - Thermal loads under thermal conditions generated by a postulated break and including T_o.

R_a - Pipe reactions under thermal conditions generated by a postulated break and including R_o.

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

Y_j - Jet impingement equivalent static load on a structure generated by a postulated break, and including an appropriate dynamic factor to account for the dynamic nature of the load.

Y_m - Missile impact equivalent static load on a structure generated by or during a postulated break, like pipe whipping, and including an appropriate dynamic factor to account for the dynamic nature of the load.

e) OTHER DEFINITIONS

S - For structural steel, S is the required section strength based on the elastic design methods and allowable stresses defined in Part 1 of the AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

U - For concrete structures, U is the section strength required to resist design loads and based on methods described in ACI 318-63, Ultimate Strength Design, Part IV-B.

Y - For structural steel, Y is the section strength required to resist design loads and based on plastic design methods described in Part 2 of AISC, "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969.

3.8.6.4.2 Load Combinations and Acceptance Criteria for Category I Concrete Structures

The following set of load combinations and allowable limits are used in evaluating and checking Category I concrete structures outside the containment for the effects of high-energy pipe breaks. Concrete barriers, used to provide a shield against the effects of high-energy pipe breaks, maintain their structural integrity under all credible loading conditions.

a) LOAD COMBINATIONS

The following load combinations should be satisfied:

1) $U = D + L + T_a + R_a + 1.5 P_a$

2) $U = D + L + T_a + R_a + 1.25 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.25$ Feqo

3) $U = D + L + T_a + R_a + 1.0 P_a + 1.0 (Y_r + Y_j + Y_m) + 1.0$ Feqs

The maximum values of P_a, T_a, R_a, Y_j, Y_r and Y_m, including an appropriate dynamic factor, shall be used unless a time-history analysis is performed to justify otherwise.

Both cases of L having its full value, possibly present during the pipe rupture event, or being completely absent are checked for.

For combinations (2) and (3), local stresses due to the concentrated loads Y_r, Y_j and Y_m, may exceed the allowables provided there will be no loss of function of any safety-related system.

3.8.6.4.3 Load Combinations and Acceptance Criteria for Category I Steel Structures

Category I steel structures outside the containment, whose function is to provide protection against the effects of high-energy pipe breaks, will have to maintain their structural integrity under all credible loading conditions. To assure this, limits on resulting stresses or required strength capacities are recommended.

a) If elastic working stress design methods are used:

1) $1.6 S = D + L + T_a + R_a + P_a$

2) $1.6 S = D + L + T_a + R_a + P_a + 1.0 (Y_j + Y_r + Y_m) +$ Feqo

3) $1.6 S = D + L + T_a + R_a + P_a + 1.0 (Y_j + Y_r + Y_m) +$ Feqs

b) If plastic design methods are used:

1) $.90 Y = D + L + T_a + R_a + 1.5 P_a$

2) $.90 Y = D + L + T_a + R_a + 1.25 P_a + 1.0 (Y_j + Y_r + Y_m) + 1.25$ Feqo

3) $.90\ Y = D + L + T_a + R_a + 1.0\ P_a + 1.0\ (Y_j + Y_r + Y_m) + 1.0\ Feqs$

In combinations 3A(a) and (b), thermal loads can be neglected when it can be shown that they are-secondary and self-limiting in nature and where the material is ductile.

In combinations (1), (2) and (3), the maximum values of P_a, T_a, R_a, Y_j, Y_r and Y_m, including an appropriate dynamic factor, are used unless a time-history analysis is performed to justify otherwise.

Both cases of L having its full value, possibly present during the pipe rupture event, or being completely absent are checked for.

For combinations (2) and (3), local stresses due to the concentrated loads Y_r, Y_j and Y_m may exceed the allowables provided there will be no loss of function. Furthermore, in computing the required section strength, S, the plastic section modulus of steel shapes may be used.

3.8.6.4.4 Acceptable Procedures for Determination of the Effect of an Impacting Whipping Pipe on Concrete and Steel Structures

Pipe whipping is precluded where the consequences would be unacceptable.

3.8.6.4.5 Acceptable Procedures for Design of Structural Pipe Restraints

Protection of Category I structures, systems and components from the dynamic effects of postulated high-energy pipe ruptures can be accomplished in some situations by providing pipe restraints in critical locations on the piping systems. These restraints should function mainly by preventing the ruptured pipe, or portions thereof, from becoming a missile that might impact and damage other critical systems, and by preventing the ruptured pipe from whipping and impacting critical systems not capable of resisting such an impact. The restraints may be independent of dead and live load supports and of seismic restraints. However, should a pipe whip restraint be intended to function also as an operating dead load and/or seismic restraint, all applicable loads should be considered in the design of the restraint.

a) <u>ANALYSIS METHOD</u>

The structural analysis of pipe restraints may consist of an energy-balance approach, where a potential collapse mechanism is first established. The displacement of this mechanism will reach its limit, by conservation of energy principles, when the external work available equals the internal work done on the restraint.

External work expressions may include kinetic expressions where mass and velocity of the ruptured pipe are known. Internal work expressions are graphically represented by the area under a resisting force-displacement curve.

b) ALLOWABLE YIELD STRENGTH

Due to the high rate of strain that the structural restraint would experience after pipe rupture, and partly due to the strain-hardening effects, the static yield strength of the material used may be increased by 15 percent.

c) ALLOWABLE STRAINS

In general, strains of up to 50 percent of ultimate strain are acceptable, provided there is no loss of function. Where buckling is critical in compression members, the load on the members should be limited to 90 percent of the buckling load.

d) GAP EFFECT

Where gaps are provided between pipes and restraints, the kinetic energy of the pipe impacting the restraint may be critical and should not be ignored. Moreover, the kinetic energy of the pipe after rebound may be more critical and should also be considered.

REFERENCES FOR SECTION 3.8

1. Kalnins, A. "Analysis of Shells of Revolution Subjected to Symmetrical and Non-Symmetrical Loads," ASME Journal of Applied Mechanics, Volume 31, Series E, No. 3.

2. Welding Research Council Bulletin 108. "A Review and Evaluation of Computer Programs for the Analysis of Stresses in Pressure Vessels".

3. Wichman, Hopper and Mershon. "Local Stresses in Spherical and Cylindrical Shells Due to External Loadings," Welding Research Council Bulletin 107.

4. Subcommittee III ACI Committee 325. "Structure Design Considerations for Pavement Joints," Journal of the American Concrete Institute, July 1956.

5. Birkeland, Philip W., and Halvard W. Birkeland. "Connections in Precast Concrete Construction," Journal of American Concrete Institute - March, 1966. Title No. 63-15.

6. Timoshenko, S., and J. H. Goodier. "Theory of Elasticity," 2nd Edition, McGraw-Hill, 1951.

7. Timoshenko, S., and S. Woinowsky - Krieger. "Theory of Plates and Shells," 2nd Edition, The Maple Press Company, York, Pa.

8. Amirikian, Arsham. "Design for Protective Structure," Bureau of Yards and Docks, Department of the Navy, August 1950.

9. "Design of Structures to Resist Nuclear Weapons Effects," ASCE Paper 42, 1961.

10. AEC letter to Florida Power & Light Company, 8-29-73, "Enclosure 2"

11. IE Bulletin 80-11 "Masonry Wall Design"

12. R. E. Uhrig (FPL) to J. P. O'Reilly (NRC) Re: RII: JPO Docket 50-335, IE Bulletin 80-11, L-81-45 dated 2/11/81.

13. R. E. Uhrig (FPL) to J. P. O'Reilly (NRC) Re: RII: JPO Docket 50-335, IE Bulletin 80-11, L-81-516 dated 12/7/81.

TABLE 3.8-1

COMPARISON OF CALCULATED DESIGN LOADS
AND ULTIMATE CAPACITY OF STRUCTURAL ELEMENTS
OF REACTOR AUXILIARY BUILDING

Structural Element	Governing Loading Condition*	Load Description	Calculated Design Value	Ultimate Capacity
Foundation Mat				
1) At support	a	Shear (kips)	3360	3600
	a	Moment(ft-kips)	250	384
2) At span centerline	a	Shear (kips)	0	3600
	a	Moment(ft-kips)	260	384
Column B2	a	Axial load (kips)	2170	2520
	a	Moment(ft-kips)	780	900
Girder 2KL - el 19.5				
1) At support	a	Shear(kips)	285	390
	a	Moment (ft-kips)	1160	1340
2) At span centerline	a	Shear (kips)	0	390
	a	Moment(ft-kips)	690	1080
South wall - el -5.0	a	Shear (kips)	19	34
	a	Moment (ft-kips)	87	224

*See Section 3.8.1.5 for load combination.

TABLE 3.8-2

COMPARISON OF CALCULATED DESIGN LOADS
AND ULTIMATE CAPACITY OF STRUCTURAL ELEMENTS
OF FUEL HANDLING BUILDING

Structural Element	Governing Loading Condition*	Load Description	Calculated Design Value	Ultimate Capacity
Foundation Mat	a	Shear(kips)	52.2	57.7
	a	Moment(ft-kips)	248	273
Column FH 4	e	Axial load(kips)	336	370
	e	Moment(ft-kips)	1774	1812
Beam FH 4				
1) At support	e	Shear(kips)	244	577
	e	Moment(ft-kips)	1406	2362
	e	Axial load(kips)	238	400
2) At span centerline	a	Shear(kips)	0	455
	a	Moment(ft-kips)	3242	3368
Wall FH 6				
1) Vertical	e	Shear(kips)	13.1	19.1
	e	Moment(ft-kips)	66.8	82.2
	e	Axial load(kips)	2.90	6.00
2) Horizontal	e	Shear(kips)	4.6	19.9
	e	Moment(ft-kips)	34.4	37.3
	e	Axial load(kips)	7.00	8.00

*See Section 3.8.1.5 for load combination

TABLE 3.8-3

COMPARISON OF CALCULATED DESIGN LOADS
AND ULTIMATE CAPACITY OF STRUCTURAL ELEMENTS
OF INTAKE STRUCTURE

Structural Element	Governing Loading Condition*	Load Description	Calculated Design Value	Ultimate Capacity
Foundation Mat	a	Shear (kips)	24	33
	a	Moment(ft-kips)	75	90
Side Wall				
a) Vertical	a	Moment(ft-kips)	64	92
b) Horizontal	a	Moment(ft-kips)	139	170
c) Punching Shear	a	Shear(kips)	840	1000
Top Slab	a	Shear(kips)	36	60
	a	Moment(ft-kips)	117	131
Strut	a	Axial load(kips)	840	1560
	a	Moment(ft-kips)	252	466

*See Section 3.8.1.5 for load combination

TABLE 3.8-4

COMPARISON OF CALCULATED DESIGN LOADS
AND ULTIMATE CAPACITY OF STRUCTURAL ELEMENTS
OF DIESEL GENERATOR BUILDING

Structural Element	Governing Loading Condition*	Load Description	Calculated Design Value	Ultimate Capacity
Foundation Mat				
a) At Support	a	Shear(kips)	25	36
	a	Moment(ft-kips)	154	160
b) At Span Center-line	a	Shear(kips)	0	36
	a	Moment(ft-kips)	115	126
External Wall-el 13.0	e	Shear (kips)	11	21
	e	Moment(ft-kips)	111	132
Roof	e	Shear (kips)	6	16
	e	Moment(ft-kips)	48	56

* See Section 3.8.1.5 for load combination

TABLE 3.8-4A

ULTIMATE HEAT SINK DAM
STABILITY ANALYSES - RESULTS OF
CRITICAL LOADING CONDITIONS

Section	Loading Combinations	Overturning Safety Factor	Sliding Safety Factor	Maximum Soil Press.	Water Levels Big mud Creek	Water Levels Intake Canal
Section I Center 33 x 50 base	DBE	1.50	1.87	-	+3.0	-7.0
	OBE	-	-	7.13 k/o'	+3.0	-7.0
	PMH	2.05	1.84	3.75 k/o'	+7.0	+16.2
Sections II Next To Center 30 x 50 base	DBE	1.51	1.56		+3.0	-7.0
		1.46	3.07		0	+3.0
	OBE	-	-	5.88 k/o'	+3.0	-7.0
	PMH	1.35	2.27	4.44 k/o'	+13.0	-6.0
Sections III Ends 15 x 28.5 base	DBE	3.75	4.66	-	+3.0	-7.0
	OBE	-	-	3.03 k/o'	+3.0	-7.0
	PMH	2.46	3.60	1.19 k/o'	+7.0	+16.2

TABLE 3.8-4B
ALLOWABLE STRESSES IN UNREINFORCED MASONRY

Description	S Allowable (psi)	S Maximum (psi)	U Allowable (psi)	U Maximum (psi)
Compressive				
Axial[(1)]	$0.22f'_m$	1000	$0.44f'_m$	2000
Flexural	$0.33f'_m$	1200	$0.85f'_m$	3000
Bearing				
On Full area	$0.25f'_m$	900	$0.62f'_m$	2250
On one-third area or less	$0.375f'_m$	1200	$0.95f'_m$	3000
Shear				
Flexural members[(2,3)]	$1.1\sqrt{f'_m}$	50	$1.7\sqrt{f'_m}$	75
Shear Walls[(2)]	$0.9\sqrt{f'_m}$	34	$1.35\sqrt{f'_m}$	51
Tension				
Normal to bed joints				
Hollow units	$0.5\sqrt{m_o}$	25	$0.83\sqrt{m_o}$	62
Solid or grouted	$1.0\sqrt{m_o}$	40	$1.67\sqrt{m_o}$	67
Parallel to bed joints[(4)]				
Hollow units	$1.0\sqrt{m_o}$	50	$1.67\sqrt{m_o}$	84
Solid or grouted	$1.5\sqrt{m_o}$	80	$2.5\sqrt{m_o}$	134
Grout Core	$2.5\sqrt{f'_c}$		$4.2\sqrt{f'_c}$	
Collar joints				
Shear		8		12
Tension		8		12

Notes:

(1) These values should be multiplied by $\left(1-\left(\frac{h}{40t}\right)^3\right)$ if the wall has a significant vertical load

(2) Use net bedded area with these stresses.

(3) For stacked bond construction use two-thirds of the values specified.

(4) For stacked bond construction use two-thirds of the values specified for tension normal to the bed joints in the head joints of stacked bond construction.

(5) Note: For St. Lucie Unit #1 m_o = 1800 psi

f'_m = 900 psi

TABLE 3.8-4C

ALLOWABLE STRESSES IN REINFORCED MASONRY

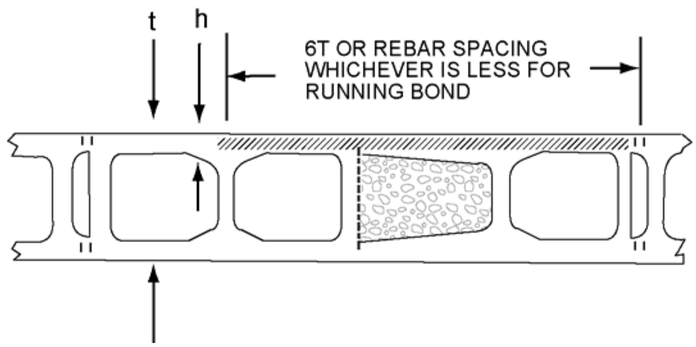
Description	S Allowable (psi)	S Maximum (psi)	U Allowable (psi)	U Maximum (psi)
Compressive				
Axial[1]	$0.22f'_m$	1000	$0.44f'_m$	2000
Flexural	$0.33f'_m$	1200	$0.85f'_m$	2400
Bearing				
On full area	$0.25f'_m$	900	$0.62f'_m$	1800
On one-third area or less	$0.375f'_m$	1200	$0.95f'_m$	2400
Shear				
Flexural members[2]	$1.1\sqrt{f'_m}$	50	$1.7\sqrt{f'_m}$	75
Shear Walls[3,4]				
Masonry Takes Shear				
$M/Vd \geq 1$	$0.9\sqrt{f'_m}$	34	$1.5\sqrt{f'_m}$	56
$M/Vd = 0$	$2.0\sqrt{f'_m}$	74	$3.4\sqrt{f'_m}$	123
Reinforcement Takes Shear				
$M/Vd \geq 1$	$1.5\sqrt{f'_m}$	75	$2.5\sqrt{f'_m}$	125
$M/Vd = 0$	$2.0\sqrt{f'_m}$	120	$3.4\sqrt{f'_m}$	180
Reinforcement				
Bond				
Plain Bars		60		80
Deformed Bars		140		186
Tension				
Grade 40		20,000		0.9Fy
Grade 60		24,000		0.9Fy
Joint Wire		.5Fy or 30,000		0.9Fy
Compression		0.4Fy		0.9Fy

TABLE 3.8-4C (Cont'd)

Notes to Table 3.8-4C:

(1) These values should be multiplied by $\left(1-\left(\frac{h}{40t}\right)^3\right)$ if the wall has a significant vertical load.

(2) This stress should be evaluated using the effective area shown in figure below except as noted in (6).



AREA ASSUMED EFFECTIVE IN FLEXURAL COMPRESSION
FORCE NORMAL TO FACE.

(3) Net bedded area shall be used with these stresses.

(4) For M/Vd values between 0 and 1 interpolate between the values given for 0 and 1.

(5) Note: For St. Lucie Unit #1 m_o = 1800 psi

f'_o = 900 psi

(6) If Dur-O-Wal reinforcement is provided for stack bond walls the effective width of the reinforced units can be increased to the same amount as that used for running bond walls.

TABLE 3.8-5

CONTAINMENT VESSEL MATERIALS

Material	Specification	Code Allow. Tensile Strength (psi)	Code	Remarks
Plate	SA 516, Gr 70 to SA 300*	17500	ASME	
Forgings	SA 350, Gr LF2*	17500	ASME	
	SA 182, F304	13750	ASME	
Pipe	SA 333,Grl*	13750	ASME	Thru 14"φ
	SB 167	18200	ASME	Thru 5"φ
	SB 166	18800	ASME	6" & 8"φ
	SA 516, Gr 70 to SA 300	17500	ASME	16"φ & Greater
Castings	SA 352, Gr LCB*	16250	ASME	
	SA 351, Gr CF8	13750	ASME	
Bolting	SA 320, Gr L43*	25000	ASME	
	SA 320, Gr B8	15000	ASME	
Structural	A 36	% Fy	AISC	Not used for pressure parts nor within 4" of pressure parts,except painter's angle.

* All of the above designated carbon steel materials comply with the requirements of the applicable ASME Code Material Specification for low temperature service, except that the impact testing, as a minimum requirement, was specified in Section III of the ASME Code, Paragraph N-1210 or N-1211 as applicable. Charpy V-Notch specimens (SA370-Type A) were used for all impact testing at a maximum temperature of 0 F.

TABLE 3.8-6

CONTAINMENT VESSEL LOAD COMBINATIONS

	LOAD COMBINATIONS										
LOAD	Case 1	Case 2	Case 3	Case 4	Case 5	Case 6	Case 7	Case 8	Case 9	Case 10	Case 11
INTERNAL PRESSURE (PSI)		49.5	44			39.6	39.6				39.6
EXTERNAL PRESSURE (PSI)				0.70	0.70						
DEAD LOAD OF VESSEL & APPUTENANCES	x	x	x	x	x	x	x	x	x	x	x
CONTAINED AIR @ TEST		x	x								
DEAD LOAD OF VENTILATION DUCT			x	x	x	x	x	x	x	x	x
DEAD LOAD OF PENETRATION INTERNALS			x	x	x	x	x	x	x	x	x
CRANE LIVE LOAD					x						
CRANE DEAD LOAD		x	x	x	x	x	x	x	x	x	x
LATERAL LOAD DUE TO WIND	x	x									
OBE HORIZONTAL LOAD				x	x	x		x		x	
DBE HORIZONTAL LOAD							x		x		x
OBE VERTICAL LOAD				x	x	x		x		x	
DBE VERTICAL LOAD							x		x		x
LIVE LOAD ON AIR LOCKS				x	x			x	x	x	
LIVE LOAD ON EQUIP.HATCHES					x						
LIVE LOAD ON PLATFORMS	x										
PIPE THERMAL LOADS								x	x	x	x
PIPE SEISMIC LOADS								x	x	x	x
PIPE RUTPURE LOADS								x	x		
THERMAL LOADS @ EMBEDMENT						x	x			x	x
JET FORCES						x	x				

TABLE 3.8-7

CONTAINMENT VESSEL

ALLOWABLE STRESS

ASME Code is used in the design of the steel shell and its penetrations. AISC refers to all other steel structures, interacting with the containment vessel, such as crane girders, platforms, and temporary supports.

CASE 1 - Construction at Post Weld Heat Treatment (PWHT)

No ASME Design (Shell is analyzed using methods consistent with the ASME Code)

AISC Design

AISC Allowables

CASE 2 - Acceptance Test at Ambient Temperature

ASME Design

$PM \leq 1.25\ (1.0\ Sm)$

$PL + PB \leq 1.25\ (1.5\ Sm)$

$PL + PB + Q \leq 3.0\ Sm$

AISC Design

AISC Allowables

CASE 3 - Pre-Operation Test at Ambient Temperature

ASME Design

$PM \leq 1.1\ (1.0\ Sm)$

$PL + PB \leq 1.1\ (1.5\ Sm)$

$PL + PB + Q \leq 3.0\ Sm$

AISC Design

AISC Allowables

CASE 4 - Normal Operating Condition at Temperature Range of 30 F to 120 F

ASME Design

$PM \leq 1.0\ Sm$

$PL + PB \leq 1.5\ Sm$

$PL + PB + Q \leq 3.0\ Sm$

AISC Design

AISC Allowables

TABLE 3.8-7 (Cont'd)

CASE 5 - Cold Shutdown at Temperature Range of 30 F to 120 F

ASME Design

$PM \leq 1.0\ Sm$ (Includes Seismic Stress)
$PL + PB \leq 1.5\ Sm$
$PL + PB + Q \leq 3.0\ Sm$

AISC Design

AISC Allowables with normal increase[1]

CASE 6 - LOCA Condition with OBE

ASME Design

$PM \leq 1.0\ Sm$ (Includes Seismic Stress)
$PL + PB \leq 1.5\ Sm$
$PL + PB + Q \leq 3.0\ Sm$

AISC Design

AISC Allowables with normal increase

CASE 7 - LOCA Condition with DBE

ASME Design

$PM \leq 0.9\ Sy$ (Includes Seismic Stress)
$PL + PB \leq 0.9\ Su$

AISC Design

AISC Allowables with normal increase

CASE 8 - Condition with OBE and Pipe Rupture

ASME Design

$PM \leq 1.0\ Sm$ (Includes Seismic Stress)
$PL + PB \leq 1.5\ Sm$
$PL + PB + Q \leq 3.0\ Sm$

NOTE:
Pipe loads are investigated as a local effect separately
$Pm \leq 0.5\ Sy$
$PL + PB \leq 1.5\ Sy$ or $1.8\ Sm$

AISC design

AISC Allowables with normal increase

TABLE 3.8-7 (Cont'd)

CASE 9 - Condition with DBE and Pipe Rupture

ASME Design

$PM \leq 0.9\ Sy$ (Includes Seismic Stress)

$PL + PB \leq 0.9\ Su$

AISC Design

AISC allowables with normal increase[1]

NOTE:
Pipe loads are investigated as a local effect separately $PM \leq 0.9\ Sy$ $PL + PB \leq 1.5\ Sy$ or $1.8\ Sm$

CASE 10 - Condition with OBE and Thermal Plus Seismic Loads on Piping

ASME Design

$PM \leq 1.0\ Sm$ (Includes Seismic Stress)

$PL + PB \leq 1.5\ Sm$

$PL + PB + Q \leq 3.0\ Sm$

AISC Design

AISC Allowables with normal increase

NOTE:
Pipe loads are investigated as a local effect separately using ASME Code allowables

CASE 11 - LOCA condition with DBE with Pressure and Thermal Plus Seismic Loads on Piping

ASME Design

$PM \leq 0.9\ Sy$

$PL + PB \leq 0.9\ SU$

AISC Design

AISC Allowables with normal increase

NOTE:
Pipe loads are investigated as a local effect separately using ASME Code allowables

Notes:

(1) "normal increase" refers to Section 1.5.6 of the AISC Code which permits a 33 1/3 percent increase in allowables for seismic stress.

TABLE 3.8-8

SUMMARY OF ELLIPSOIDAL BOTTOM HEAD STRESSES

	MERIDIONAL (LONGITUDINAL) STRESSES				CIRCUMFERENTIAL STRESSES			
	COMPRESSIVE STRESS (PSI)		TENSILE STRESS (PSI)		COMPRESSIVE STRESS (PSI)		TENSILE STRESS (PSI)	
LOAD CASE	ALLOWABLE	MAX. CALC.	ALLOWABLE	MAX. CALC.	ALLOWABLE	MAX CALC.	ALLOWABLE	MAX. CALC.
1	-	0	5000	166	833	185	5000	0
2	-	0	21875	20928	-	0	21875	18384
3	-	0	19250	18618	-	0	19250	16324
4	-	0	17500	72	2200	261	17500	0
5	-	0	17500	72	2200	261	17500	0
6	-	0	17500	16769	-	0	17500	14677
7	-	0	30670	16769	-	0	30670	14677
8	-	0	17500	161	-	0	17500	180
9	-	0	34200	161	-	0	34200	180

TABLE 3.8-9

SUMMARY OF CYLINDER STRESSES

	MERIDIONAL (LONGITUDINAL) STRESSES				CIRCUMFERENTIAL STRESSES			
	COMPRESSIVE STRESS (PSI)		TENSILE STRESS (PSI)		COMPRESSIVE STRESS (PSI)		TENSILE STRESS (PSI)	
LOAD CASE	ALLOWABLE	MAX. CALC.	ALLOWABLE	MAX. CALC.	ALLOWABLE	MAX. CALC.	ALLOWABLE	MAX. CALC.
1	3000	1030	5000	222	305	111	5000	111
2	-	0	21875	10781	-	0	21875	21891
3	-	0	19250	9547	-	0	19250	19360
4	3900	1257	17500	0	850	110	17500	0
5	3900	1307	17500	0	850	110	17500	0
6	-	0	17500	8626	-	0	17500	17424
7	-	0	30670	8770	-	0	30670	17424
8	3900	1202	17500	0	-	0	17500	0
9	3900	1620	34200	56	-	0	34200	0

TABLE 3.8-10

SUMMARY OF HEMISPHERICAL DOME STRESSES

	MERIDIONAL (LONGITUDINAL) STRESSES				CIRCUMFERENTIAL STRESSES				
	COMPRESSIVE STRESS (PSI)		TENSILE STRESS (PSI)		COMPRESSIVE STRESS (PSI)		TENSILE STRESS (PSI)		
LOAD CASE	ALLOWABLE*	MAX. CALC.	ALLOWABLE	MAX. CALC.	ALLOWABLE	MAX. CALC.	ALLOWABLE	MAX. CALC.	51
1	790	235	5000	24	790	192	5000	328	
2	-	0	21875	21653	-	0	21875	22095	
3	-	0	19250	19230	-	0	19250	19487	
4	2036	364	17500	0	2036	212	17500	74	
5	2036	364	17500	0	2036	212	17500	74	
6	-	0	17500	17316	-	0	17500	17608	
7	-	0	30670	17342	-	0	30670	17664	
8	2036	254	17500	0	2036	102	17500	184	
9	2036	310	24200	0	2036	123	34200	240	

* Meridional Stress + Circumferential Stress ≤ 790 psi or 2036 psi as applicable.

TABLE 3.8-10A

GUARD PIPE AND PROCESS LINE ASSEMBLY

ACTUAL VS. ALLOWABLE STRESSES

Penetration Type	Pen No.	Pen Function	Process Pipe - Flued Head Weld		Guard Pipe - Flued Head Weld		Guard Pipe Pipe Rupture + Operating Stress (Allowable (YIA0))
			OBE + Operating Weld Stress-psi (Allowable 3(SM))	DBE + Operating Weld Stress-psi (Allowable 3(SM))	OBE + Operating Weld Stress-psi (Allowable 3(SM))	DBE + Operating Weld Stress (Allowable 3(SM))	
I	1	Main Steam	24,672 (54,921)	43,446 (54,921)	12,474 (62,844)	22,602 (62,843)	14,200 (23,400)
I	3	Feedwater	15,636 (58,682)	18,679 (58,682)	6,575 (62,210)	9,360 (63,210)	23,400 (30,800)
III	5	Blowdown	21,644 (57,124)	28,683 (56,739)	1,605 (60,000)	2,082 (60,000)	11,200 (27,100)
III	26	Letdown	8,351 (51,022)	8,439 (51,022)	11,327 (59,487)	11,360 (59,488)	6,530 (27,100)
III	36	Safety-Injection	19,520 (57,464)	25,522 (47,464)	14,840 (60,000)	16,060 (60,000)	13,800 (31,400)
III	40	Shutdown	25,770 (56,100)	33,684 (56,100)	11,397 (56,100)	12,238 (56,100)	14,700 (33,100)

TABLE 3.8-10B

FLUID HEAD MAXIMUM STRESS SUMMARY
FOR
TYPE I PENETRATIONS
SEISMIC LOADINGS

Main Steam (Pl, P2)

Combination	Max. Stress Intensity (PSI)	Stress Location	Allowable Stress
#1 OBE&THERM(Ax+Bend+Tor)	26203	El#90-Head Body I.D. Left Side	61141
#2 OBE&THERM(Ax+Trans+Bend+Tor)	36732	El#82 O.D.Left Process Pipe Hub	61080
#3 OBE&THERM(Ax+Trans+Bend+Tor)	25806	El#90 Head Body I.D. Left Side	61141
#4 DBE&THERM(Axial)	26110	El#90 Head Body I.D. Left Side	61141
#5 DBE&THERM(Ax+Trans+Bend+Tor)	25950	El#90 Head Body I.D. Left Side	61141
#6 DBE&THERM(Ax+Trans+Bend+Tor)	57626	El#114 O.D. Left Process Pipe Hub	61442
#7 DBE&THERMAL(Trans+Bend+Tor)	25338	El#90 Head Body ID Left Side	61141

Feedwater (P3, P4)

Combination	Max. Stress Intensity (PSI)	Stress Location	Allowable Stress
#1 OBE&THERM(Ax+Trans+Bend+Tor)	32205	El#100 Head Body I.D. Left Side	60520
#2 OBE&THERM(Ax+Trans+Bend+Tor)	34783	El#101 Head Body I.D. Left Side	60473
#3 OBE&THERM(Ax+Trans+Bend+Tor)	31736	EI#100 Head Body I.D. Left Side	60520
#4 DBE&THERM(Ax+Trans+Bend+Tor)	32984	El#101 Head Body I.D. Left Side	60473
#5 DBE&THERM(Ax+Trans+Bend+Tor)	38188	El+101 Head Body I.D. Left Side	60473
#6 DBE&THERM(Ax+Trans+Bend+Tor)	31944	El#100 Head Body I.D. Left Side	60520

TABLE 3.8-10C

FLUID HEAD MAXIMUM STRESS SUMMARY
FOR
TYPE III PENETRATIONS
SEISMIC LOADINGS

PENETRATION NUMBER 5

Combination	Max. Stress Intensity(PSI)	Stress Location	Allowable Stress
#1 O.B.E. Normal OP.(Ax+BEN)	23776	OD Outboard Process Pipe Hub(#64)	57729
#2 O.B.E. Normal OP.(Trans+BEN)	19845	OD Inboard Process Pipe Hub (#77)	57124
#3 O.B.E. Normal OP (Trans+Tor)	24575	OD Outboard Process Pipe Hub (#64)	57124 57729
#4 D.B.E. EMER.(Ax+BEN)	31158	OD Outboard Process Pipe Hub (#64)	57729
#5 D.B.E. EMER.(Trans+BEN)	26940	OD Inboard Process Pipe Hub (#77)	57124
#6 D.B.E. EMER.(Trans+Tor)	35689	OD Outboard Process Pipe Hub (#64)	57729

PENETRATION NUMBER 26

Combination	Max. Stress Intensity (PSI)	Stress Location	Allowable Stress
#1 O.B.E. Normal OP.(Ax+BEN)	25588	Inboard Process Pipe Fillet RAD (#183)	54188
#2 O.B.E. Normal OP.(Trans+BEN)	25608	"	"
#3 O.B.E. Normal OP.(Trans+Tor)	25616	"	"
#4 D.B.E. EMER.(Ax+BEN)	25581	"	"
#5 D.B.E. EMER.(Trans+BEN)	25608	"	"
#6 D.B.E. EMER.(Trans+Tor)	25620	"	"

TABLE 3.8-10C (Cont'd)

PENETRATION NUMBER 36

Combination	Max. Stress Intensity (PSI)	Stress Location	Allowable Stress
O.B.E. #1 Normal OP.(Ax+Ben)	19036	ID Head(#125)	57852
O.B.E. #2 Normal OP.(Trans+Ben)	19118	ID Head(#125)	57852
O.B.E. #3 Normal OP.(Trans+Tor)	20402	Outboard Process Pipe Hub (#23)	57744
D.B.E. #4 Emer.(Ax+Ben)	23569	OD Inboard Process Pipe Hub (#91)	57508
D.B.E. #5 Emer.(Trans+Ben)	22365	OD Inboard Process Pipe Hub (#91)	57508
D.B.E. #6 Emer.(Trans+Tor)	26775	Outboard Process Pipe Hub (#23)	57744

Max Transient Stress = 25430(ALT)@505 cycles /Usage Factor <.0l
Stress located at I.D. of Outboard Process Pipe Hub

PENETRATION NUMBER 40

Combination	Max. Stress Intensity (PSI)	Stress Location	Allowable Stress
O.B.E. #1 Normal OP.(Ax+Ben)	22542	ID Inboard Process Pipe Hub (#86)	56100
O.B.E. #2 Normal OP.(Trans+Ben)	22300	OD Outboard Process Pipe Hub (#22)	"
O.B.E. #3 Normal OP.(Trans+Tor)	22439	"	"
D.B.E. #4 Emer.(Ax+Ben)	25945	"	"
D.B.E. #5 Emer.(Trans+Ben)	25918	"	"
D.B.E. #6 Emer. (Trans+Tor)	29889	"	"

TABLE 3.8-11

COMPARISON OF CALCULATED DESIGN LOADS

AND ULTIMATE CAPACITY OF STRUCTURAL ELEMENTS
OF SHIELD BUILDING

Structural Element	Governing Loading Condition*	Load Description	Calculated Design Value	Ultimate Capacity
Foundation Mat				
a) Radial	g	Shear (kips/ft)	26	124
	g	Moment(ft-kips)	1730	2000
b) Tangential	g	Shear(kips/ft)	22	124
	g	Axial load(kips/ft)	253	260
	g	Moment (ft-kips/ft)	1478	1792
Cylinder Wall				
a) Vertical below el 10.0	g	Shear (kips/ft)	11	73
	g	Axial load (kips/ft)	68	80
	g	Moment (ft-kips/ft)	246	312
b) Horizontal below el 10.0	g	Shear(kips/ft)	3	73
	g	Axial load (kips/ft)	69	80
	g	Moment (ft-kips/ft)	42	147
c) Vertical above el 10.0	g	Shear (kips/ft)	0.5	40
	g	Axial load (kips/ft)	51	55
	g	Moment (ft-kips/ft)	56	101
d) Horizontal above el 10.0	g	Shear (kips/ft)	0.7	40
	g	Axial load (kips/ft)	115	120
	g	Moment (ft-kips/ft)	60	155

*See Section 3.8.2.2.6 for load combination.

TABLE 3.8-11 (Cont'd)

Structural Element	Governing Loading Condition*	Load Description	Calculated Design Value	Ultimate Capacity
Dome				
a) Meridional	i	Shear (kips/ft)	19	32
	i	Axial load (kips/ft)	17	30
	i	Moment (ft-kips/ft)	85	333
b) Circumferential	i	Shear(kips/ft)	19	32
	i	Axial load (kips/ft)	214	242
	i	Moment (ft-kips/ft)	13	13
Secondary Shield Wall				
a) Vertical	h	Shear (kips/ft)	171	171
	h	Axial load (kips/ft)	136	137
	h	Moment (ft-kips/ft)	634	687
b) Horizontal	h	Shear (kips/ft)	179	179
	h	Axial load (kips/ft)	404	405
	h	Moment (ft-kips/ft)	578	598
Primary Shield Wall				
a) Vertical	h	Shear (kips/ft)	35	144
	h	Moment(ft-kips/ft)	1336	1360
b) Horizontal	h	Moment (ft-kips/ft)	310	336

*See Section 3.8.2.2.6 for load combination.

TABLE 3.8-12

COATING - INSIDE CONTAINMENT

DATA INTENTIONALLY DELETED

TABLE 3.8-12 (con't)

REPAIR COATING SCHEDULE - INSIDE CONTAINMENT

DATA INTENTIONALLY DELETED

TABLE 3.8-13
RESTRAINT ANALYSIS - LINES ANALYZED
(Historical Information)

Fluid In Pipe	System & Line Number	Fluid in Pipe	System & Line Number	Fluid in Pipe	System & Line Number
Steam	Main Steam:	Water	Reactor Coolant:	Water	Reactor Coolant:
"	MS-1,3	"	RC-114,123	"	RC-834
"	MS-2,4	"	RC-112,115,121,124	"	RC-885,836
"	MS-10,11,13	"	RC-108	Water	Chemical & Volume Control System:
"	MS-28,29	"	RC-147,151 to 154,162	"	CH-106,107,110,111
"	MS-50,51	"	RC-102,103	"	CH-104,109,112,125
"	MS-52,53	"	RC-109,141	"	CH-146,147,148,149
"	MS-63 to 78	"	RC-113,116,122,125	"	CH-126,127,135,136,137
“	M-79,80	“	RC-142,145,148.149		
Water	Feedwater:	"	RC-150	"	CH-113,117,121
"	BF-28,30	"	RC-105,117,118.119	"	CH-115,118,123,134
"	BF-42,43	"	RC-120,130,137,138	"	CH-101,102
"	BF-32	"	RC-139,140	"	CH-100,103,128
"	BF-33,35	"	RC-101	"	CH-142,143
"	Deleted	"	RC-156,157	"	CH-300,301,312
"	BF-13,18,55,56	"	RC-104,107	"	CH-309,310
"	BF-14,19	"	RC-822	"	CH-304,307
"	BF-29,34,31,36	"	RC-824,827,828,829	"	CH-305,308
"	BF-51,52	"	RC-825,826		

TABLE 3.8-13 (Cont'd)

Fluid in Pipe	System & Line Number	Fluid in Pipe	System & Line Number
Water	Containment Spray & Safety Injection:	Water	Blowdown System
"	CS-8,9,10,11	"	I-B-42,43,61,62
"	CS-14,15,18,19	"	B-52,54
"	SI-406,407,412,414		
"	SI-408,410,814		
"	SI-472,474		
"	SI-430		
"	SI-415,416		
"	CS-58,59		
"	CS-36,37		
"	SI-222,224		
"	CS-38,39		
"	SI-110,111,112,113		
"	SI-137,138,139,140		
"	SI-457,458,459,460		
"	SI-101,102,103,100, 148,149,150,151		
	SI 1" lines at Safety Injection Tank		

Refer to drawing
8770-G-793 Sheet 1

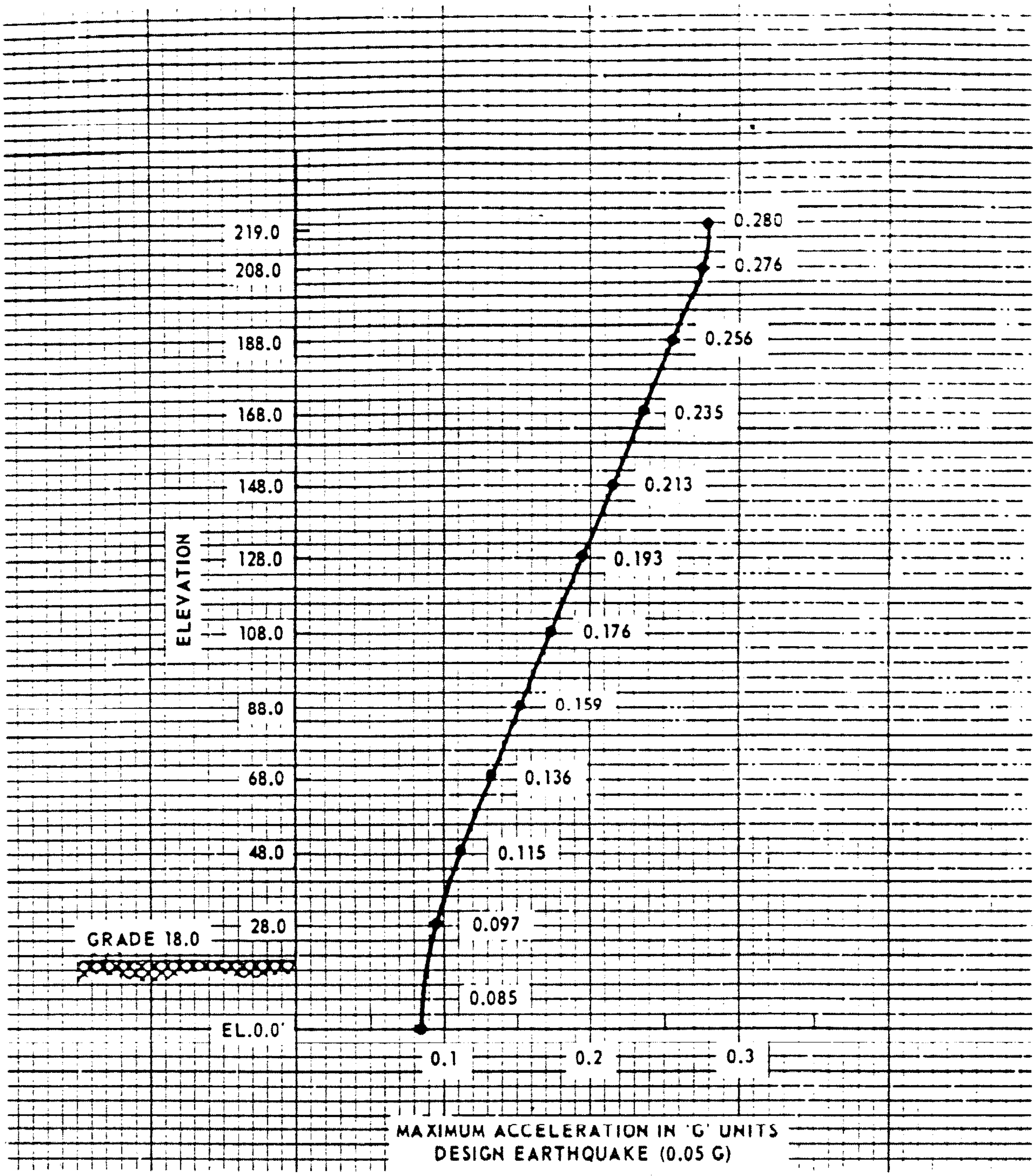
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

CONTAINMENT VESSEL SH.1

FIGURE 3.8-1

Refer to drawing
8770-G-793 Sheet 4

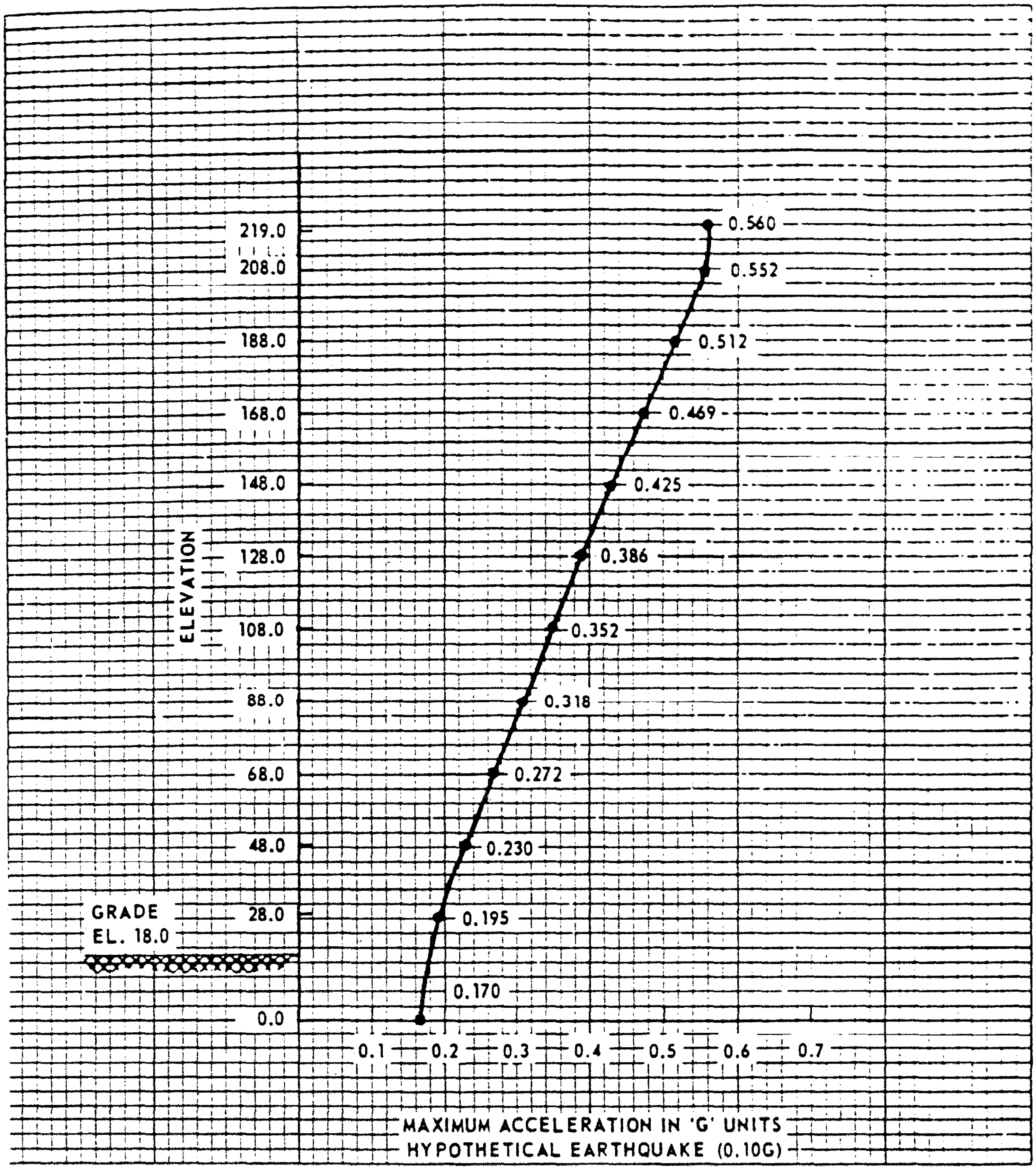
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
CONTAINMENT VESSEL SH.4 FIGURE 3.8-1a



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

CONTAINMENT VESSEL OBE SEISMIC LOADING - SHEET 1

FIGURE 3.8-2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

CONTAINMENT VESSEL OBE SEISMIC
LOADING - SHEET 2

FIGURE 3.8-3

Refer to drawing
8770-G-493

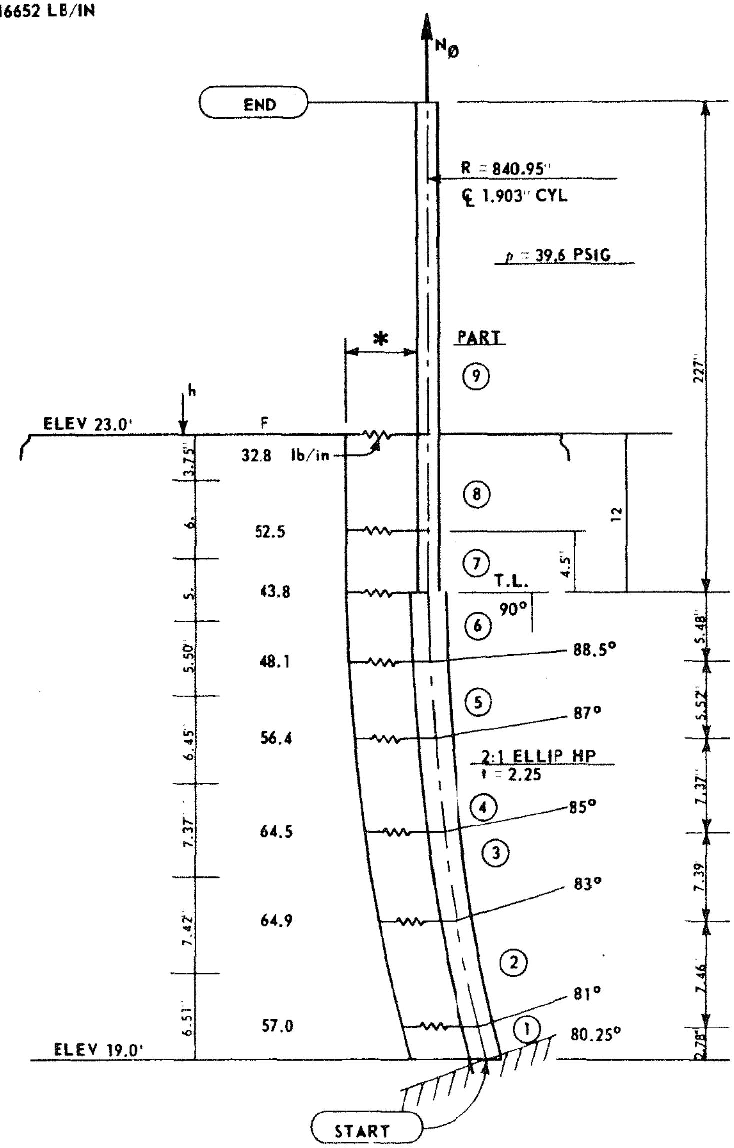
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING CONTAINMENT VESSEL
SUPPORT PLAN & SECTIONS M & R
FIGURE 3.8-3a

BOUNDARY CONDITIONS

AT END
$Q = 0$
$N_{\phi} = \frac{pr}{2} = \frac{39.6\ (840.95)}{2} = 16652$ LB/IN
$M_{\phi} = 0$

AT START
$W = 0$
$U_{\phi} = 0$
$B_{\phi} \equiv 0$



* 4" THICK DOWN CHEM CO. ETHAFOAM 220

ASSUMING 10% COMPRESSION
$E = 35$ PSI $= \frac{3.5 \text{ PSI STRESS}}{.10 \text{ STRAIN}}$

$k = \frac{35}{4} = 8.75$ PSI/IN

SPRING LOAD PER UNIT WIDTH OF CIRCUM,

$F = 3.75\ (8.75)\ (1) = kh$
$F = 32.8$ LB/IN

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

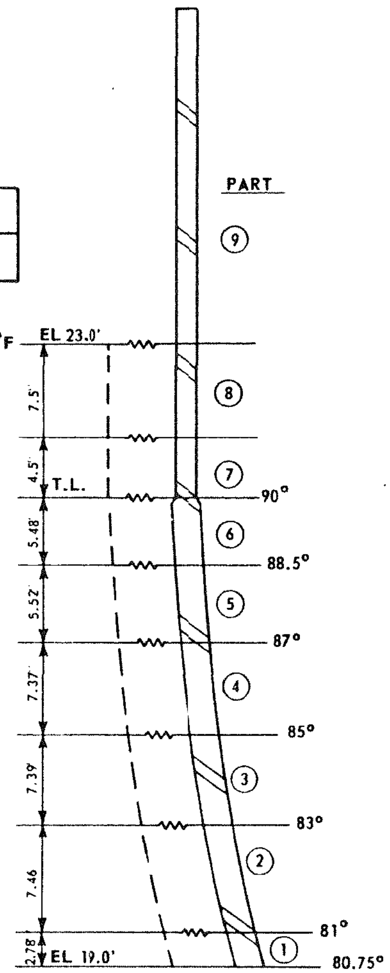
CONTAINMENT VESSEL EMBEDMENT STRESS MODEL

FIGURE 3.8-4

SHELL TEMPERATURES UNDER
ACCIDENT CONDITIONS

(120°F AMBIENT BEFORE ACCIDENT)

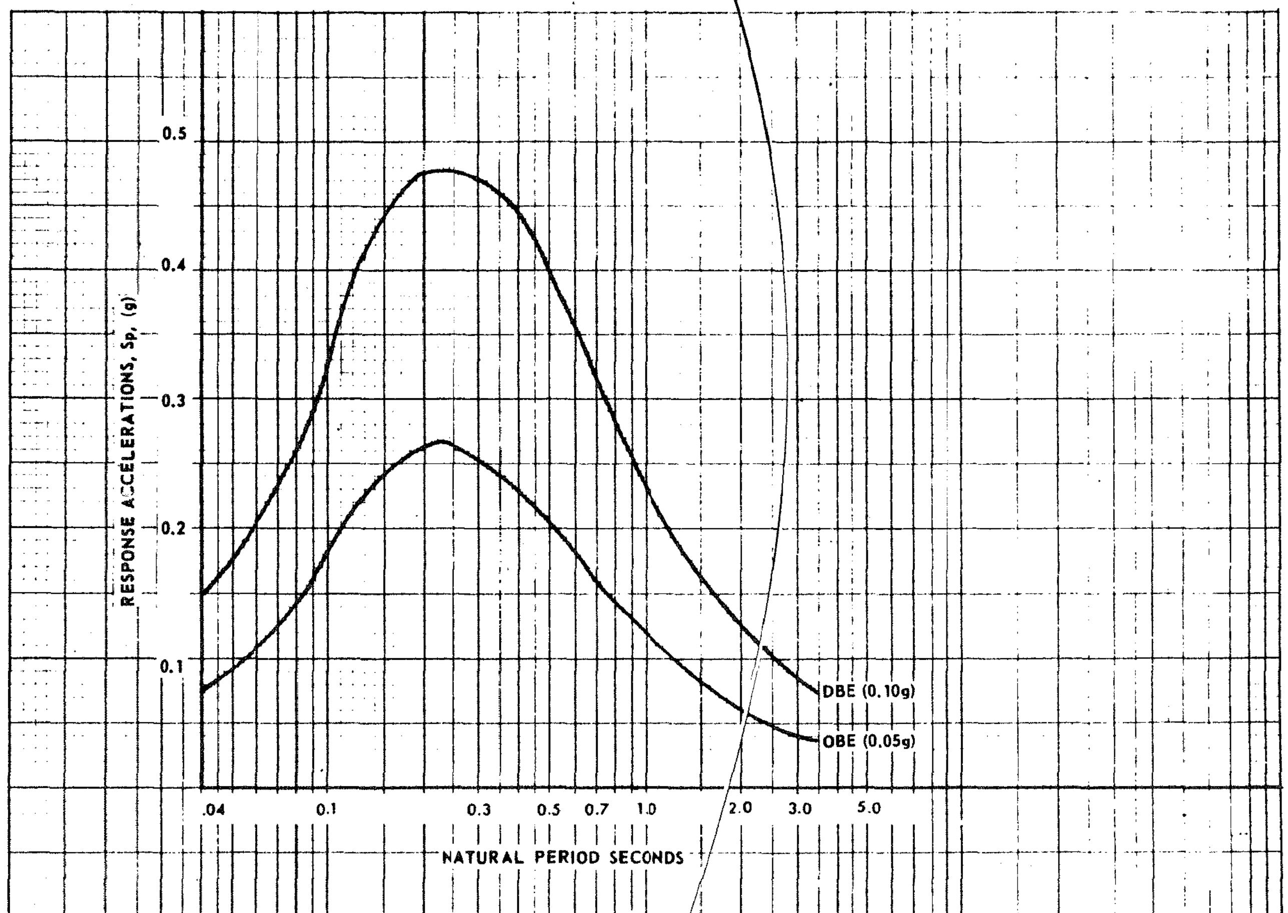
CASE	1	4	7	10
TIME	5 MIN	20 MIN	35 MIN	50 MIN
	205°F	241°F	247°F	249°F
	205	241	247	249
	205	241	247	249
	156	183	197	205
	125	151	163	171
	120	129	135	141
	120	121	125	128
	120	120	122	123
	120	120	121	122



FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

EMBEDMENT LOCA TEMPERATURE GRADIENT

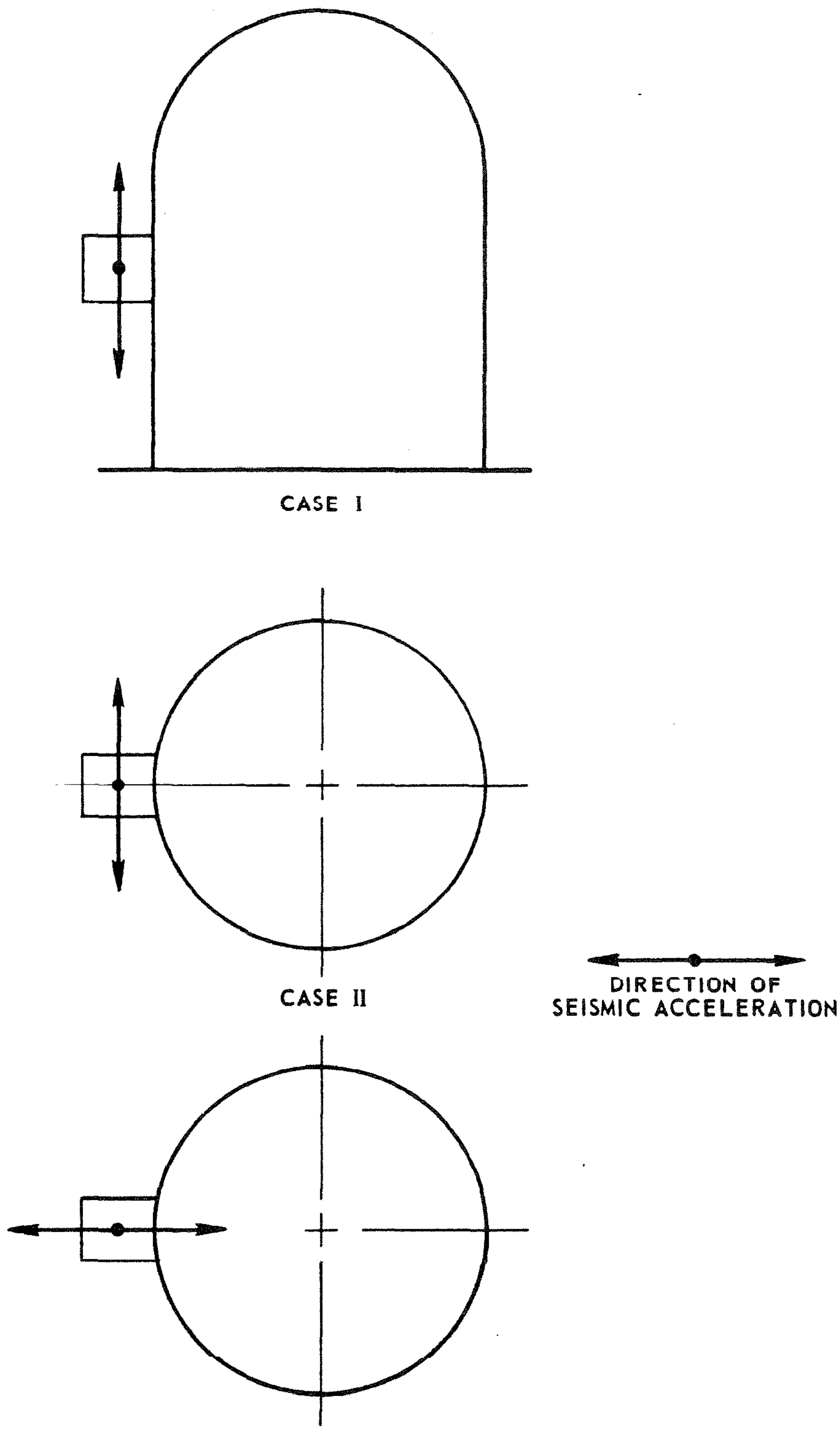
FIGURE 3.8-5



FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

AIR LOCK SEISMIC RESPONSE SPECTRA

FIGURE 3.8-6



FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

AIRLOCK SEISMIC ANALYSIS MODEL

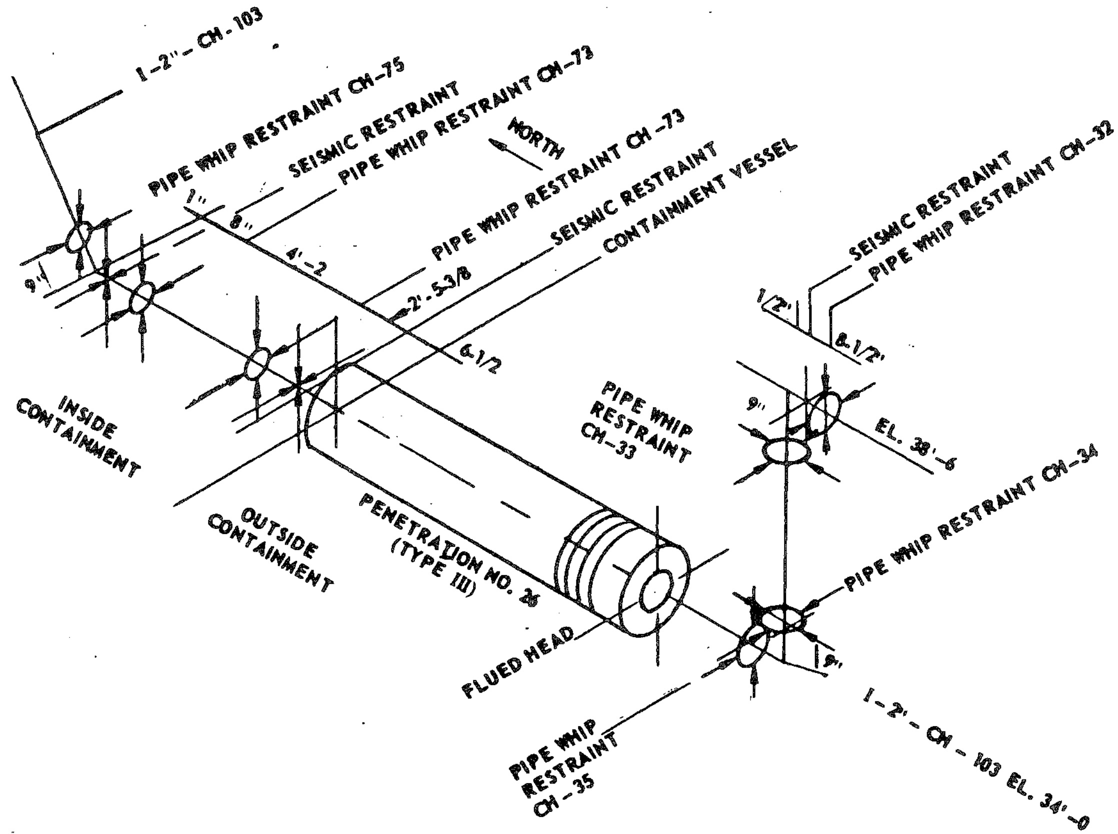
FIGURE 3.8-7

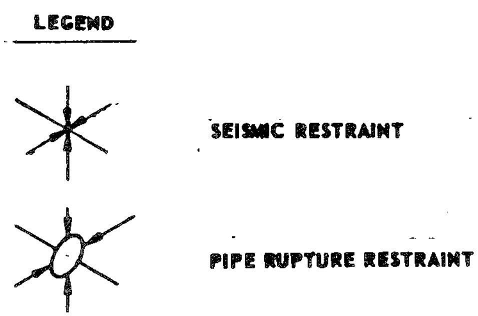
Security Related Information
Figure Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

ELECTRICAL PENETRATIONS

FIGURE 3.8-8





NOT TO SCALE

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

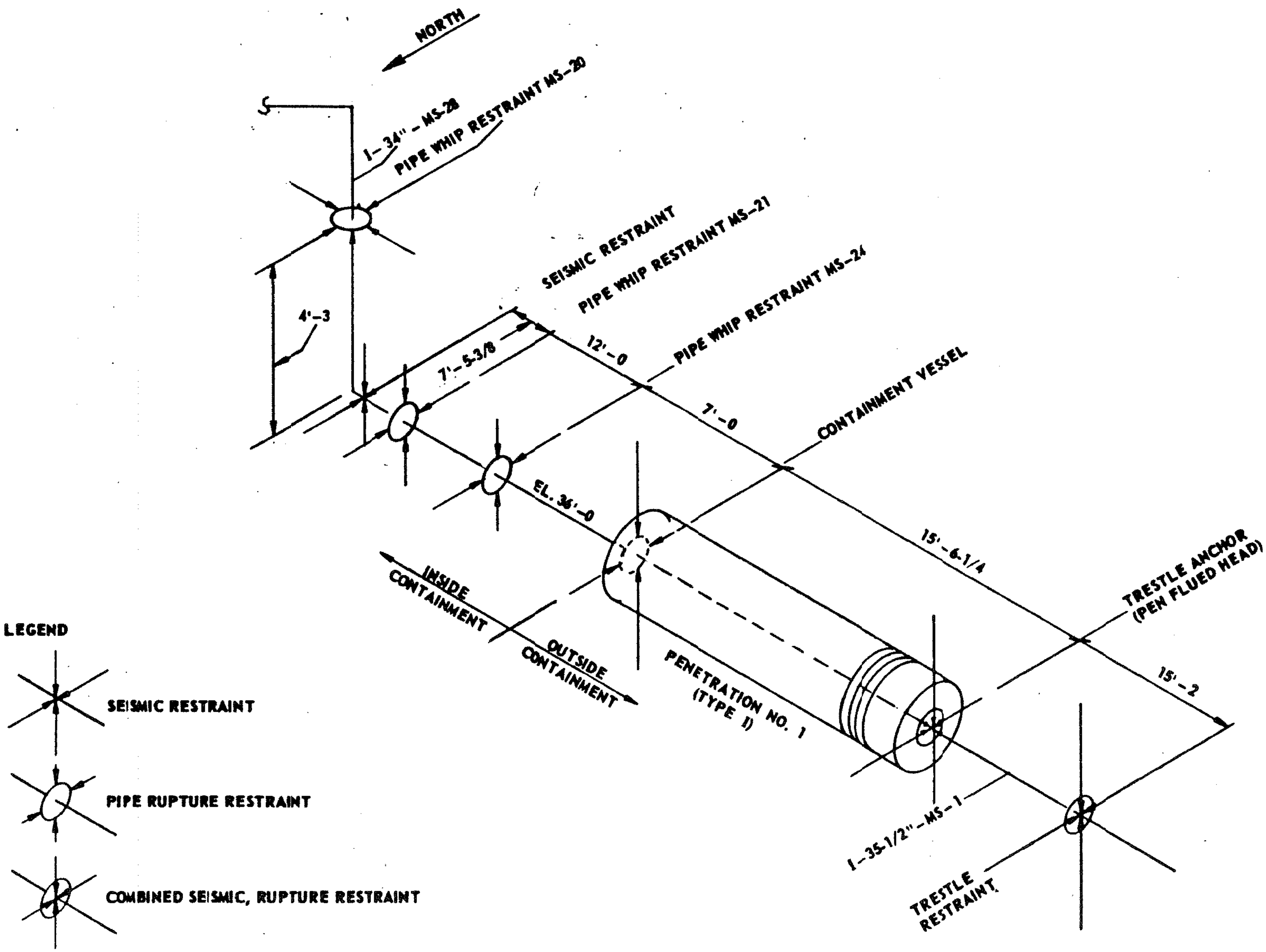
LETDOWN PENETRATION 26 ADJACENT RESTRAINTS

FIGURE 3.8-8A

Refer to drawing
8770-G-814 Sheet 9

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MISCELLANEOUS STRUCTURAL STEEL ALL AREAS SH.9
FIGURE 3.8-8aa



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MAIN STEAM PENETRATION 1
ADJACENT RESTRAINTS

FIGURE 3.8-8B

Refer to drawing

8770-G-213 Sheet 3

FLORIDA POWER AND LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS SH. 3 FIGURE 3.8-9

Refer to drawing

8770-G-213 Sheet 4

FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS
(SHIELD WALL PIPE SLEEVES)
FIGURE 3.8-9a

Refer to drawing
8770-G-213 Sheet 4

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS
FIGURE 3.8-9a

Refer to drawing

8770-G-213 Sheet 1

FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR CONTAINMENT BUILDING PIPING
PENETRATIONS SH. 1
FIGURE 3.8-10

Refer to drawing

8770-G-213 Sheet 2

FLORIDA POWER AND LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REACTOR CONTAINMENT BUILDING PIPING PENETRATIONS SH. 2 FIGURE 3.8-11

Refer to drawing
8770-G-793 Sheet 2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

CONTAINMENT VESSEL SH.2

FIGURE 3.8-12

Refer to drawing
8770-G-793 Sheet 3

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

CONTAINMENT VESSEL SH.3

FIGURE 3.8-13

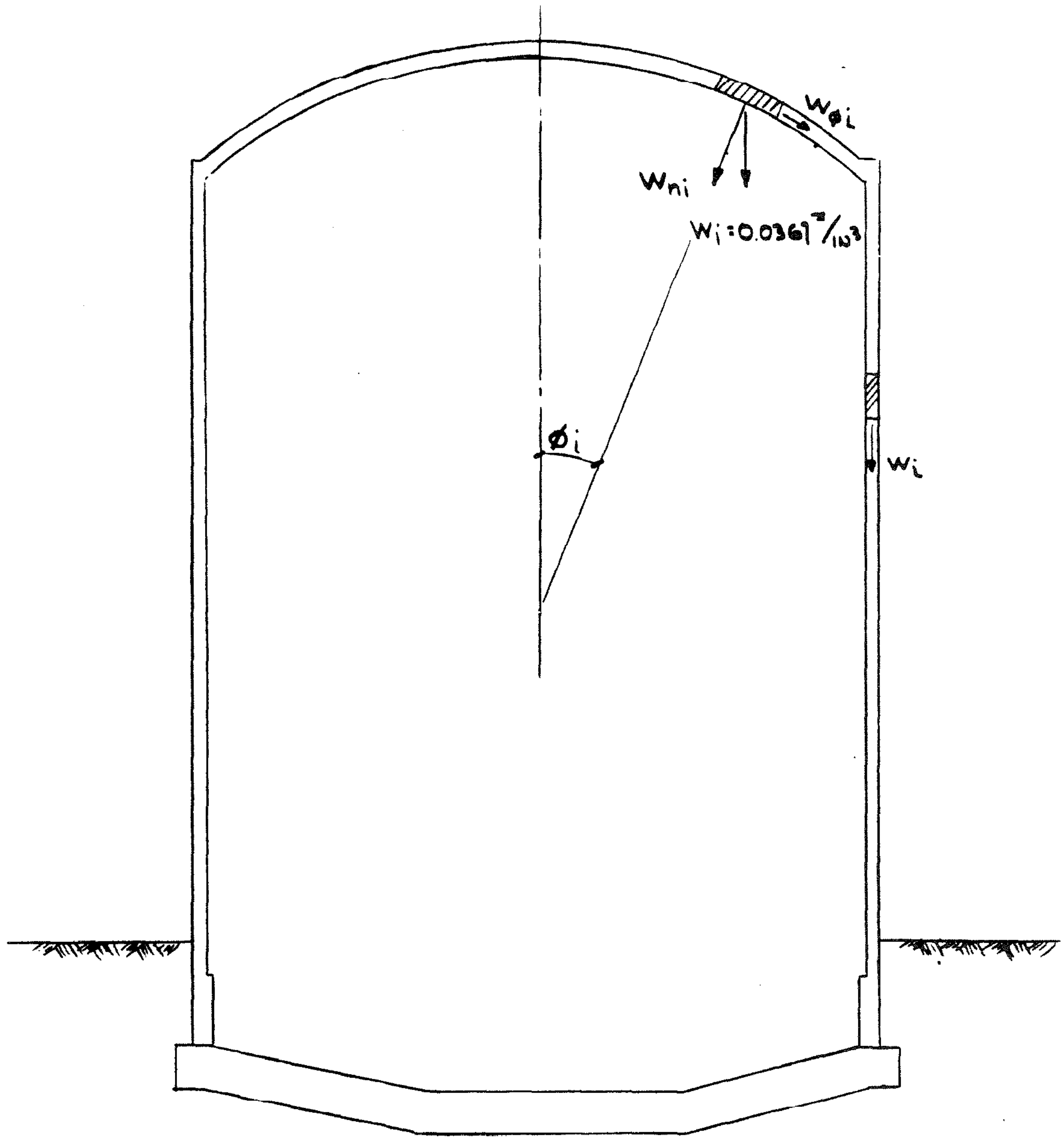
Security Related Information
Figure Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

SHIELD BUILDING
COMPUTING MODEL LAYOUT

FIGURE 3.8-14



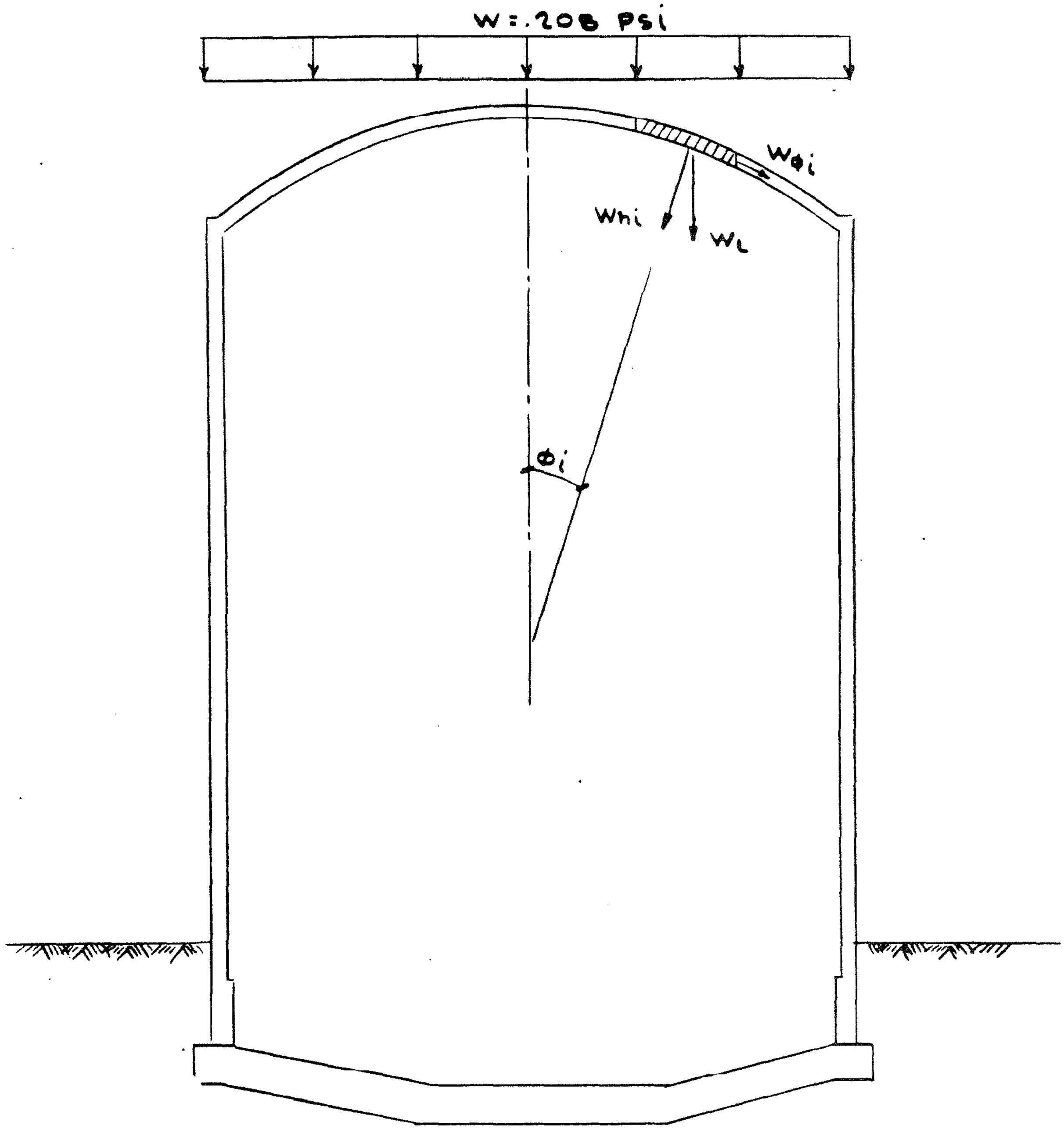
LOADING CONDITION I DEAD LOAD

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

SHIELD BUILDING
DEAD LOAD CONDITION

FIGURE 3.8-15



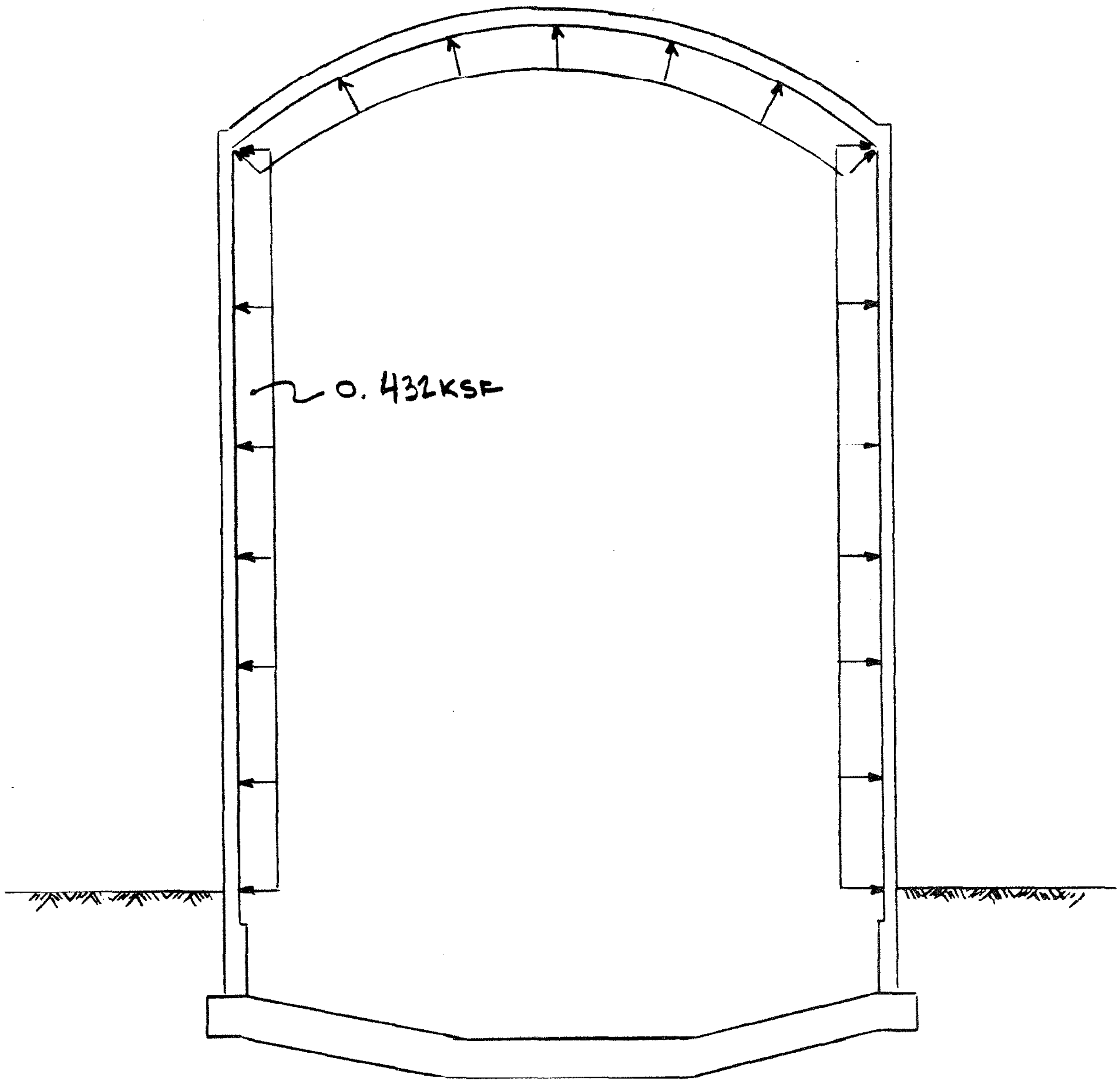
LOADING CONDITION 2 LIVE LOAD

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

SHIELD BUILDING
LIVE LOAD CONDITION

FIGURE 3.8-16



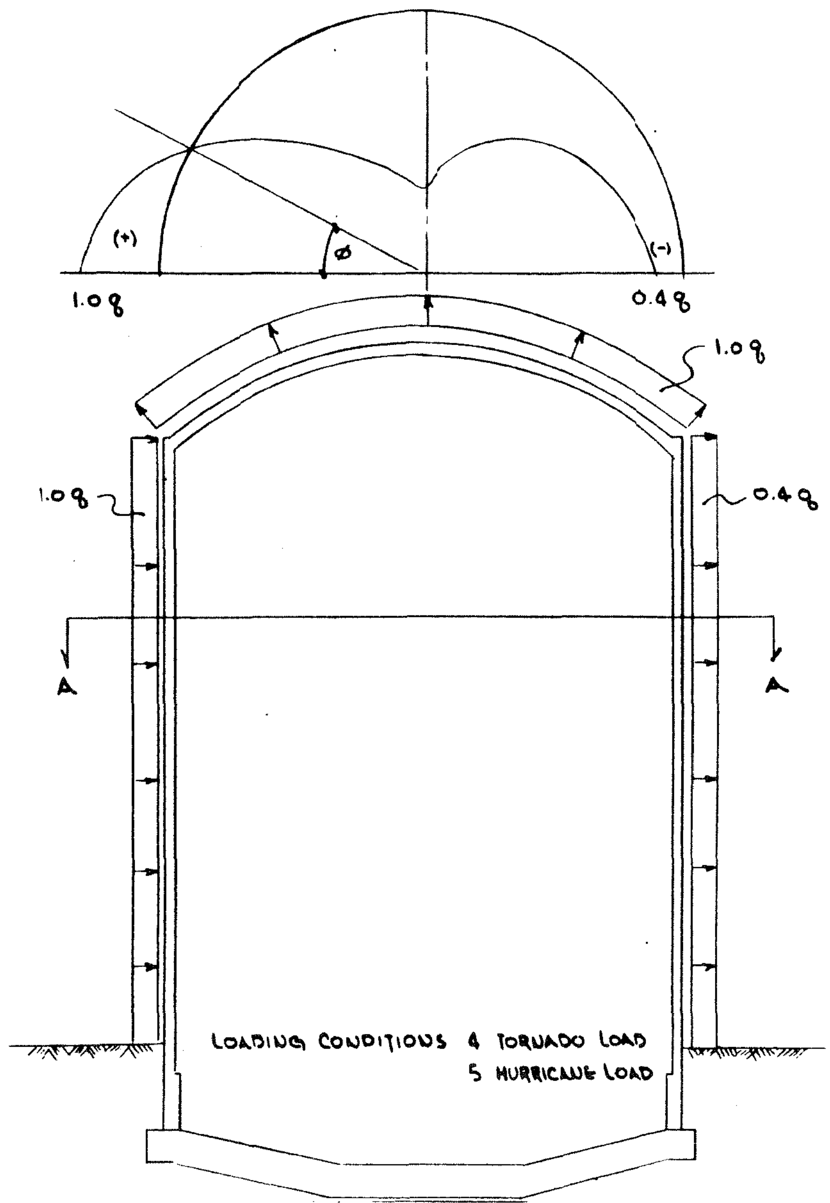
LOADING CONDITION 3 UNIFORM PRESSURE - 3.0 PSI

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

SHIELD BUILDING TORNADO INTERNAL PRESSURE LOAD CONDITION

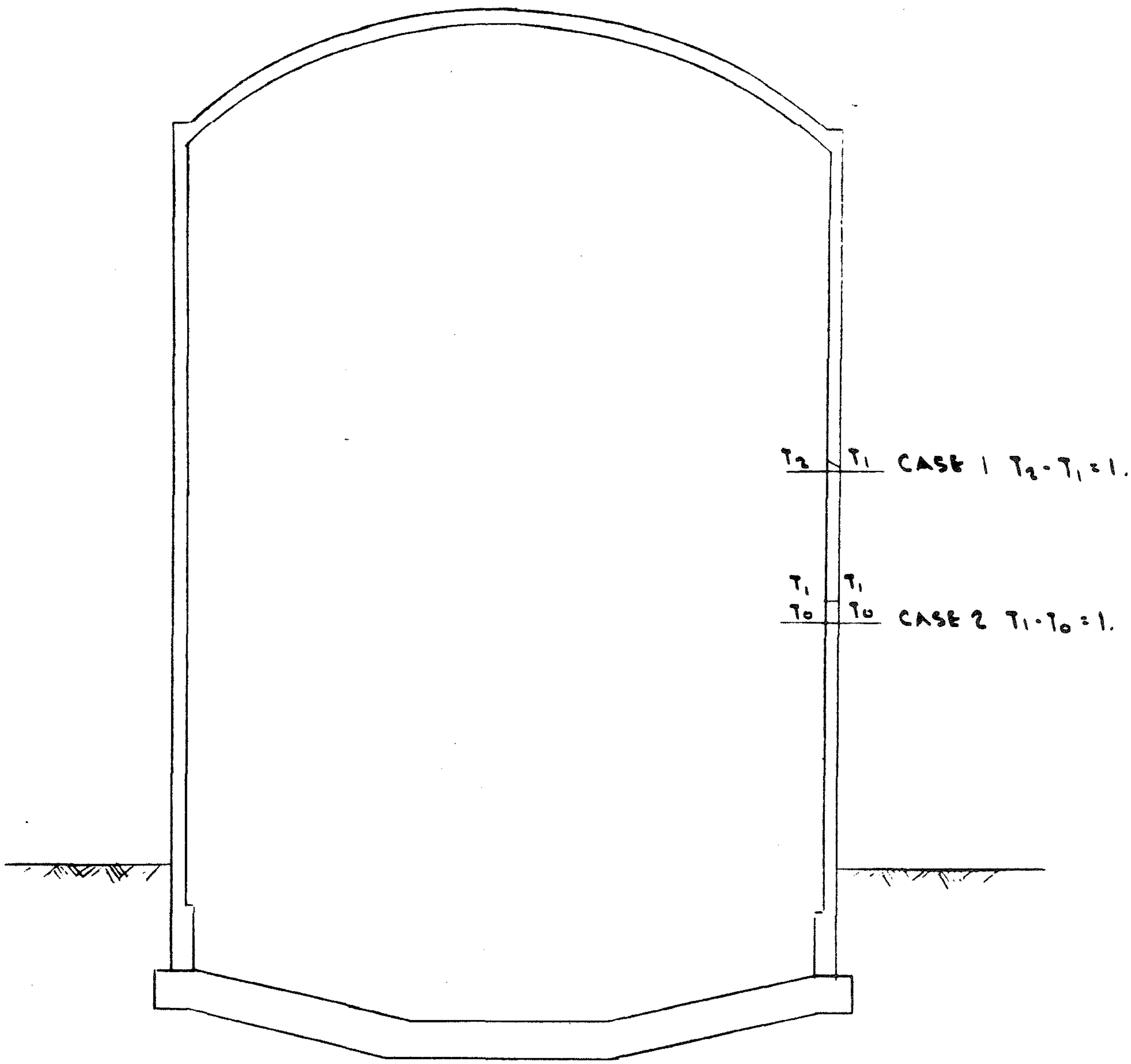
FIGURE 3.8-17



FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

SHIELD BUILDING TORNADO AND HURRICANE WIND LOAD CONDITION

FIGURE 3.8-18



LOADING CONDITION TEMPERATURE LOADS

CASE 1 - UNIT GRADIENT

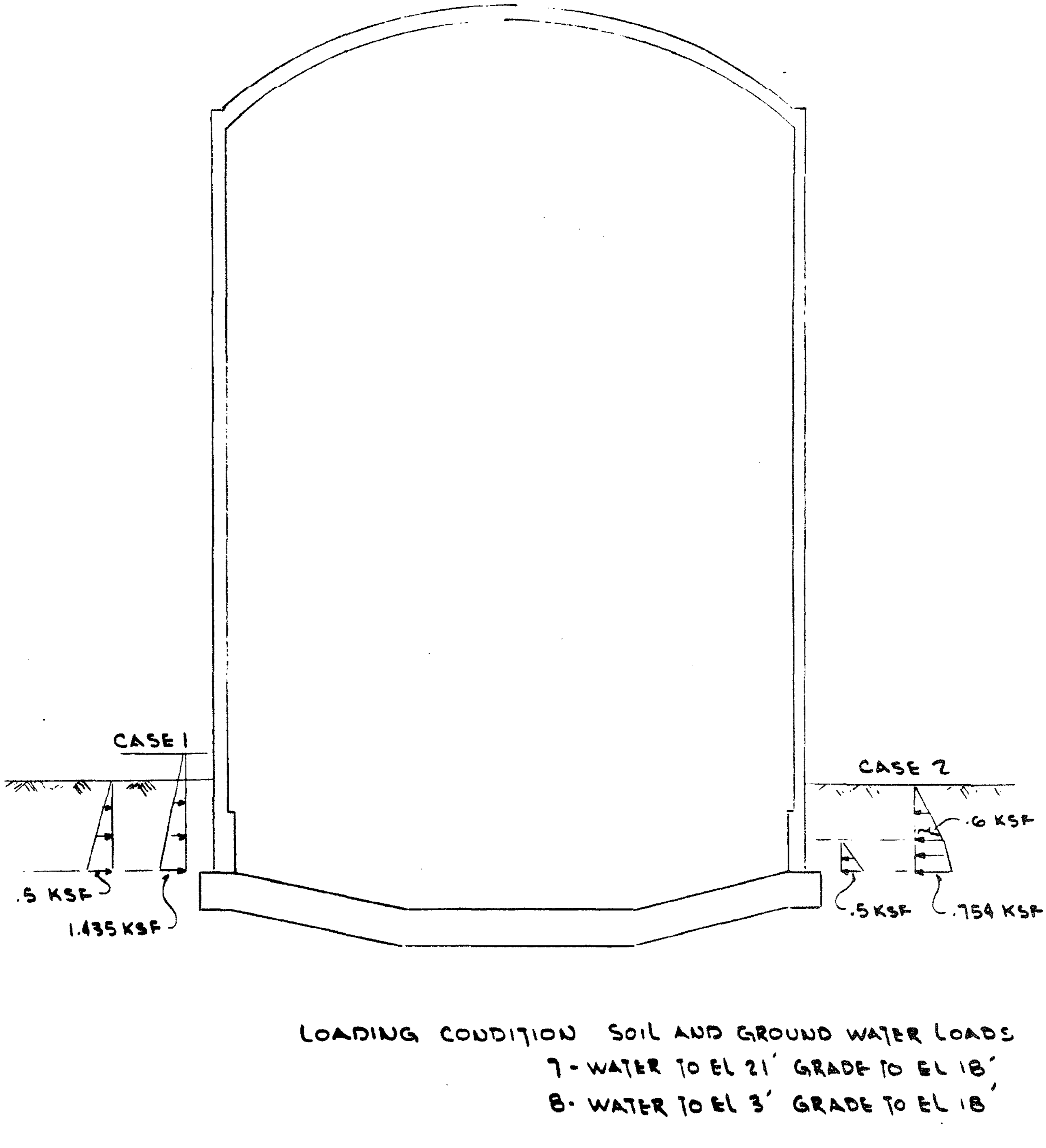
CASE 2 - UNIT HOMOGENEOUS INCREASE

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

SHIELD BUILDING
TEMPERATURE LOAD CONDITION

FIGURE 3.8-19

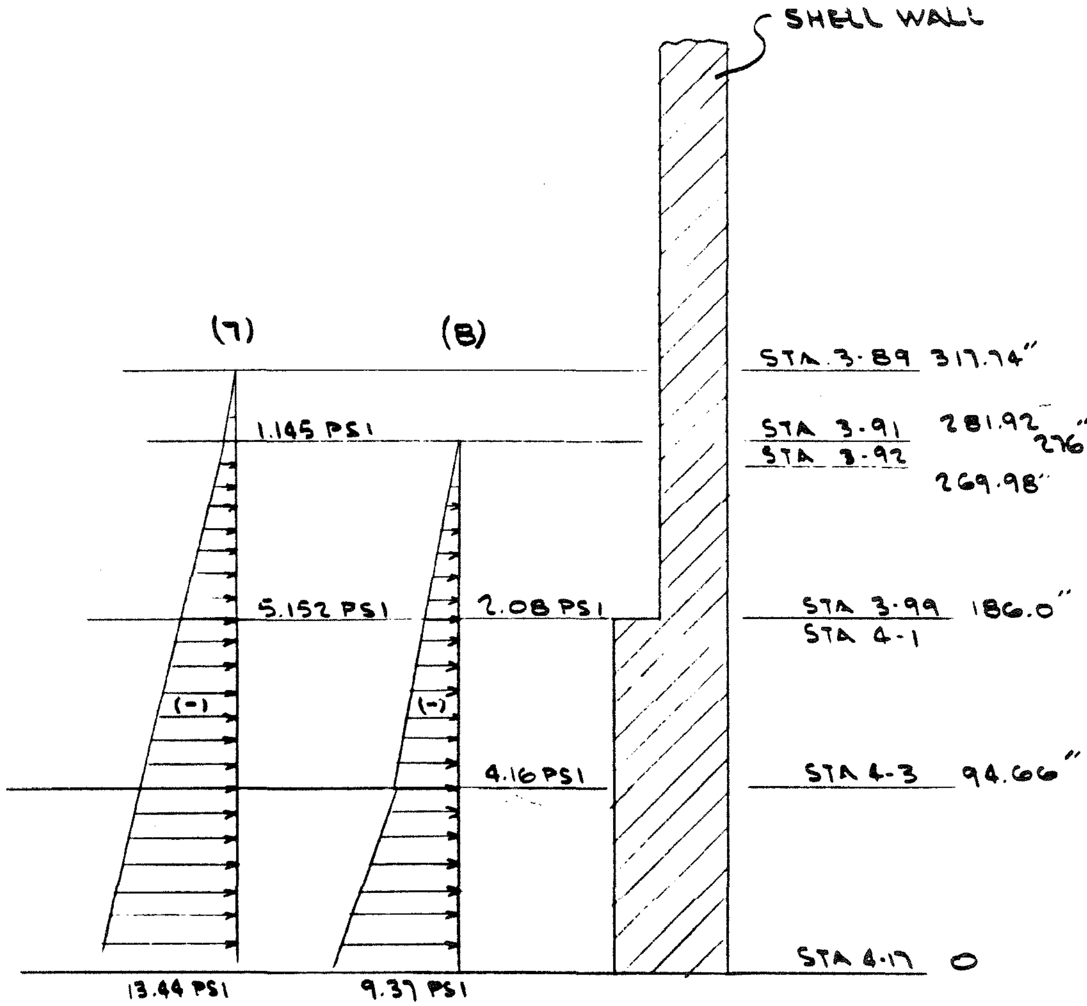


FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

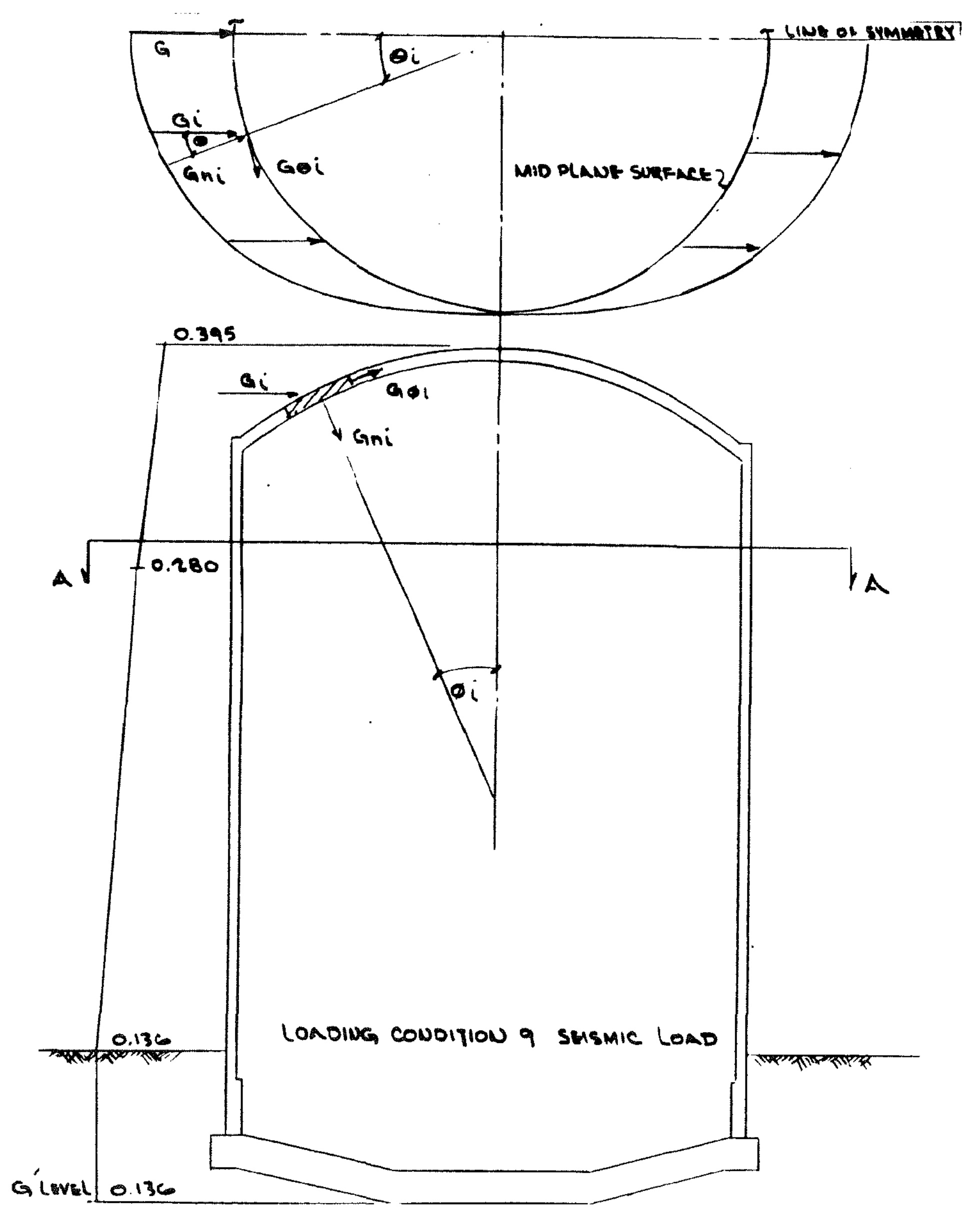
SHIELD BUILDING
SOIL AND GROUND WATER
LOAD CONDITIONS

FIGURE 3.8-20



LOAD CONDITIONS (7)(8) - WATER AND EARTH PRESSURES

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT
SHIELD BUILDING WATER AND EARTH PRESSURE PROFILE
FIGURE 3.8-21



FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

SHIELD BUILDING SEISMIC LOAD CONDITION

FIGURE 3.8-22

Refer to drawing
8770-G-488

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING BASE SLAB-PLAN-MAS

FIGURE 3.8-23

Refer to drawing
8770-G-490

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING BASE SLAB-PLAN REINF-
SH. NO.1
FIGURE 3.8-24

Refer to drawing
8770-G-501

FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REACTOR BUILDING CYLINDER WALL-PLAN & SECT-MAS FIGURE 3.8-25

Refer to drawing
8770-G-503

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING CYLINDER DEV. MAS.

FIGURE 3.8-26

Refer to drawing
8770-G-511

FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REACTOR BUILDING DOME-PLAN & SECTS.- MAS FIGURE 3.8-27

Refer to drawing
8770-G-512

FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REACTOR BUILDING DOME-REINF SH 1 FIGURE 3.8-28

Refer to drawing
8770-G-513

FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REACTOR BUILDING DOME-REINF. SH 2 FIGURE 3.8-29

Refer to drawing
8770-G-518

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

REACTOR BUILDING INTERNAL CONC. PLANS & SECTS.-MAS.-SH. NO.1
FIGURE 3.8-30

Refer to drawing
8770-G-519

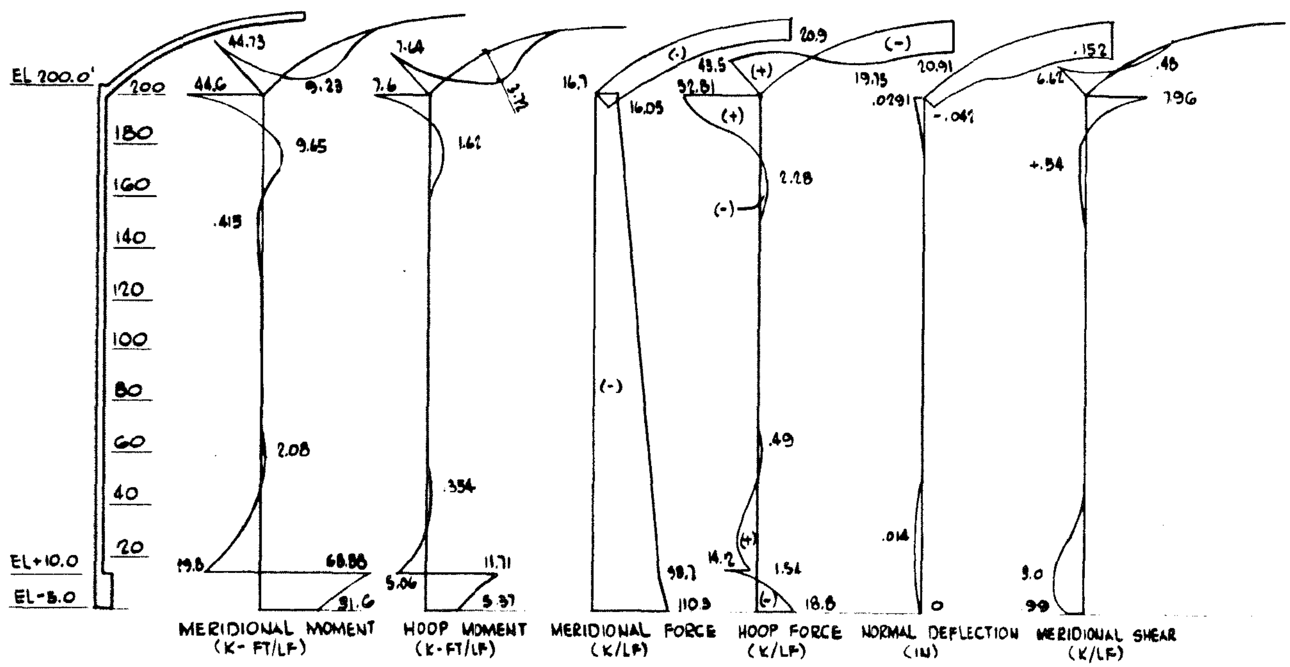
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REACTOR BUILDING INTERNAL CONC. PLANS & SECTS.-MAS.-SH. NO.2 FIGURE 3.8-31

Refer to drawing

8770-G-803

FLORIDA POWER AND LIGHT COMPANY ST. LUCIE PLANT UNIT 1
REACTOR BUILDING EMBEDDED STEEL- PRIMARY SHIELD WALL FIGURE 3.8-31a

DEAD LOAD 150#/CF



NOTE:

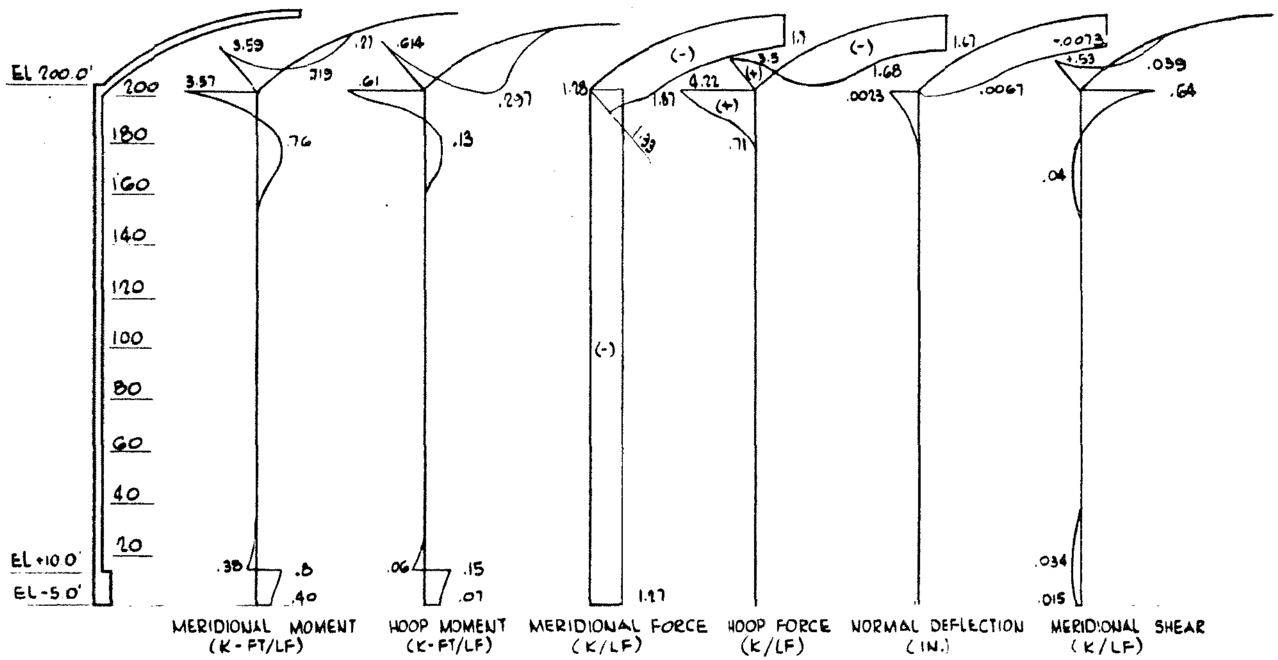
MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF STRUCTURAL ANALYSIS DEAD LOAD

FIGURE 3.8-32

LIVE LOAD 30#/SF VERTICAL



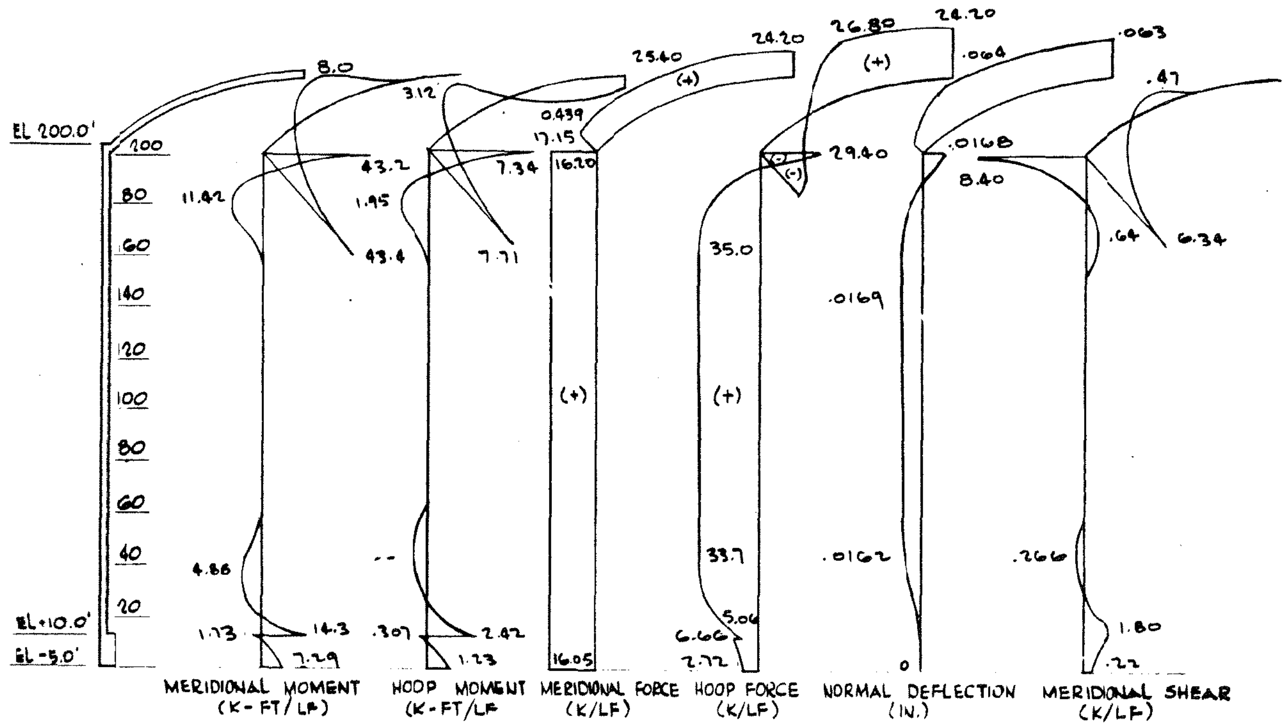
NOTE:

MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING
STRUCTURAL ANALYSIS
LIVE LOAD
FIGURE 3.8-33

UNIFORM INTERNAL PRESSURE = 3.00 PSI



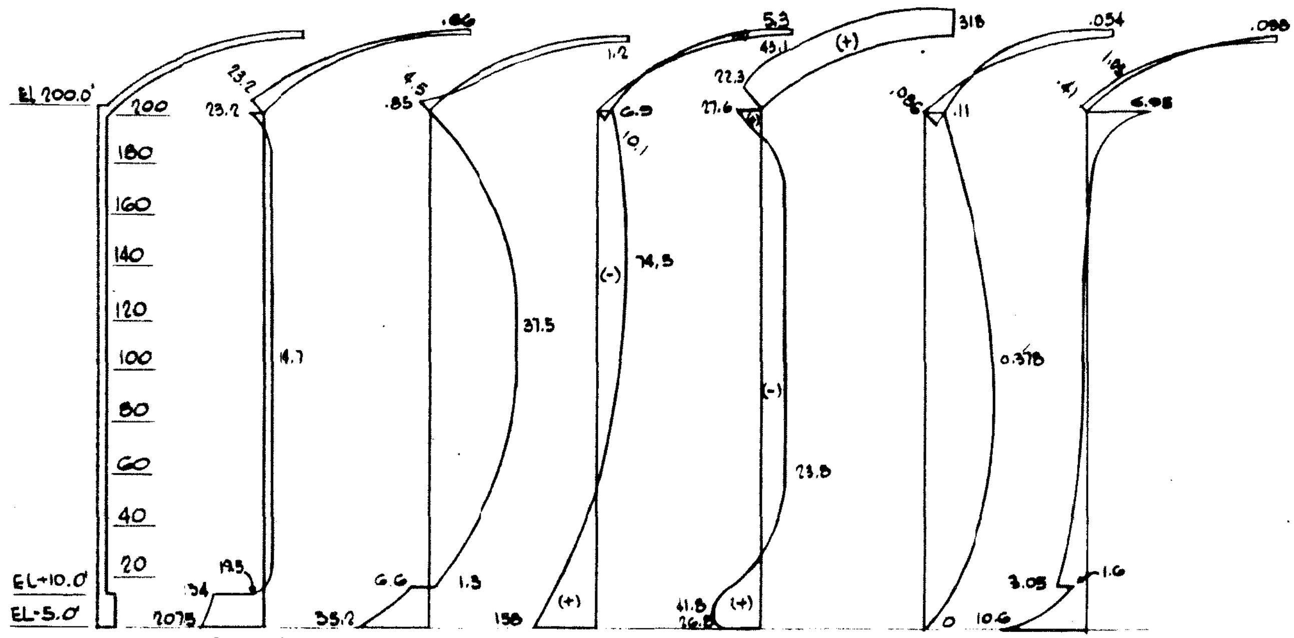
NOTE:

MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING STRUCTURAL ANALYSIS
TORNADO INTERNAL PRESSURE LOAD
FIGURE 3.8-34

1.0 TORNADO (360 MPH) θ = 0°



MERIDIONAL MOMENT (K-FT/LF)　HOOP MOMENT (K-FT/LF)　MERIDIONAL FORCE (K/FT)　HOOP FORCE (K/FT)　NORMAL DEFLECTION (IN)　MERIDIONAL SHEAR (K/FT)

NOTE:

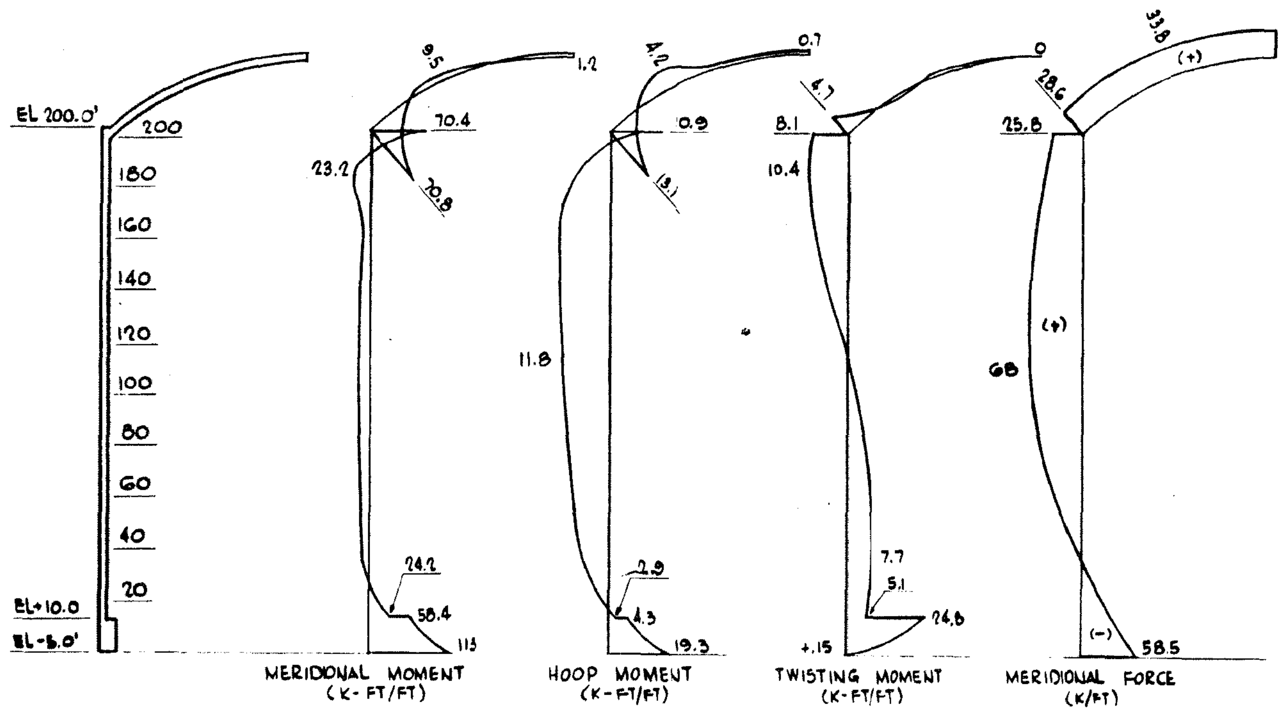
MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING
STRUCTURAL ANALYSIS
TORNADO WIND LOAD – 0°

FIGURE 3.8-35

1.0 TORNADO (360 MPH) θ = 90°



NOTE:

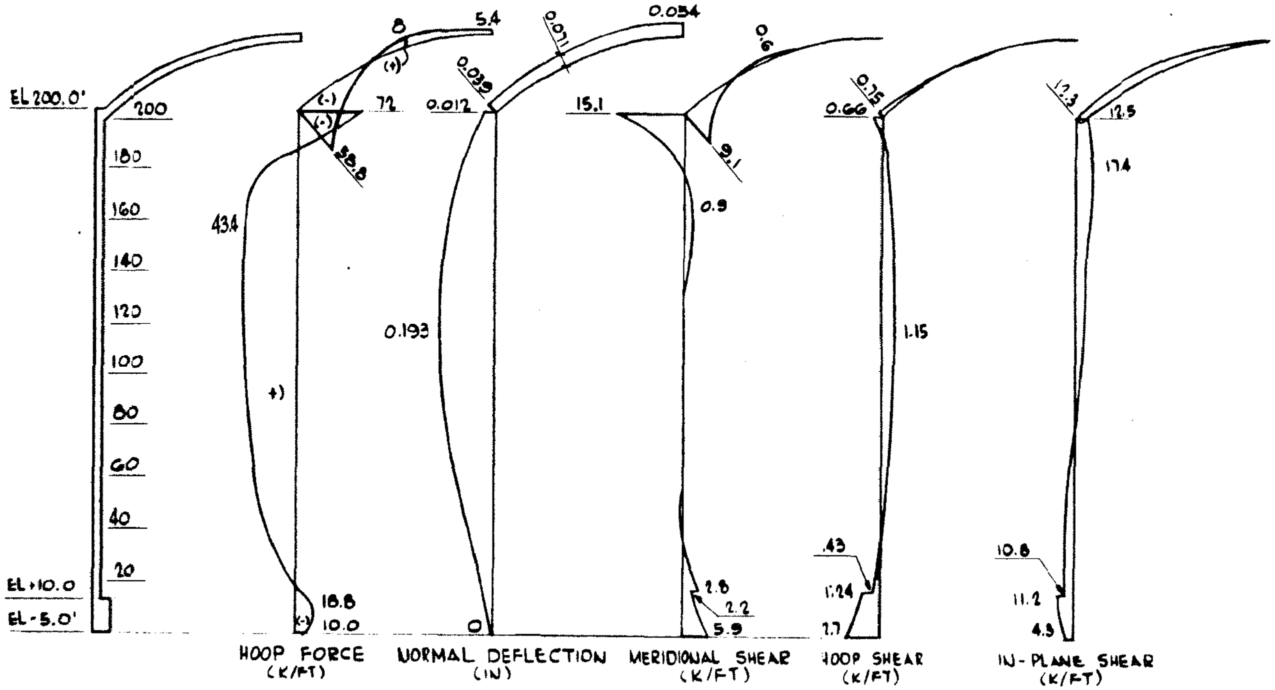
MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING STRUCTURAL ANALYSIS TORNADO WIND LOAD – 90°
SHEET 1

FIGURE 3.8-36

1.0 TORNADO (360 MPH) θ = 90°



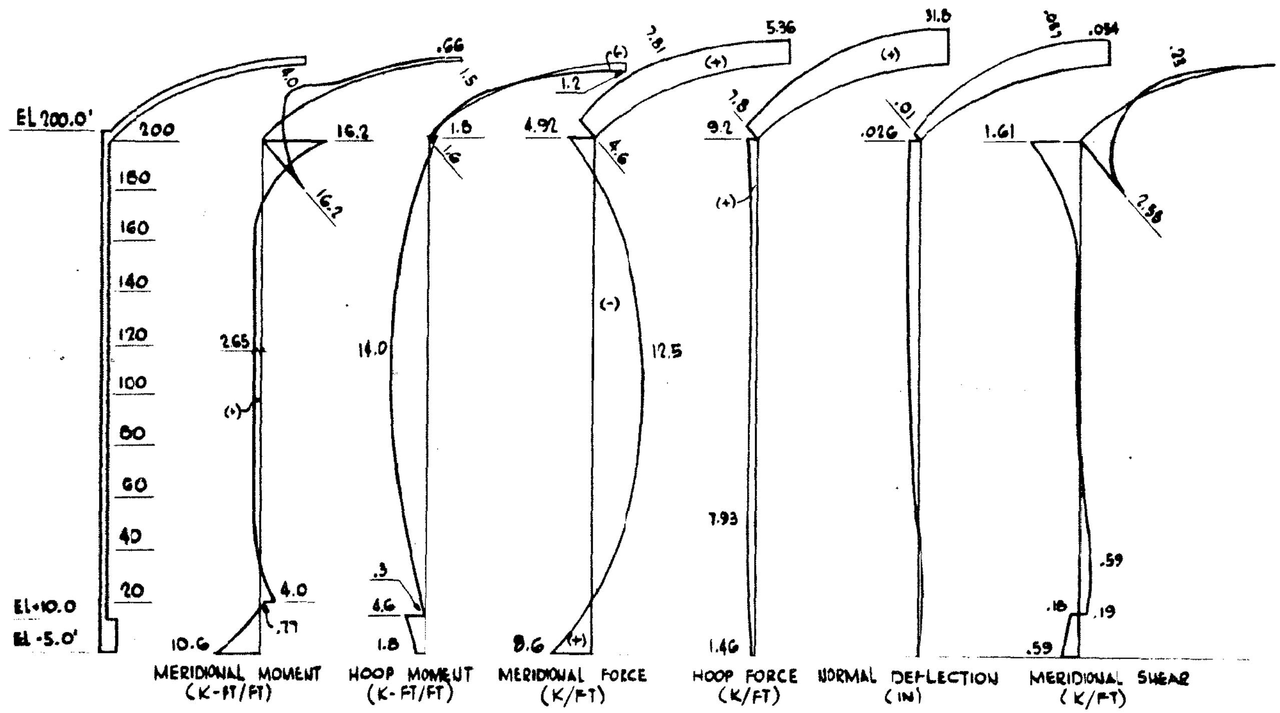
NOTE:

MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING STRUCTURAL ANALYSIS TORNADO WIND LOAD - 90°
SHEET 2
FIGURE 3.8-37

1.0 TORNADO (360 MPH) θ = 180°



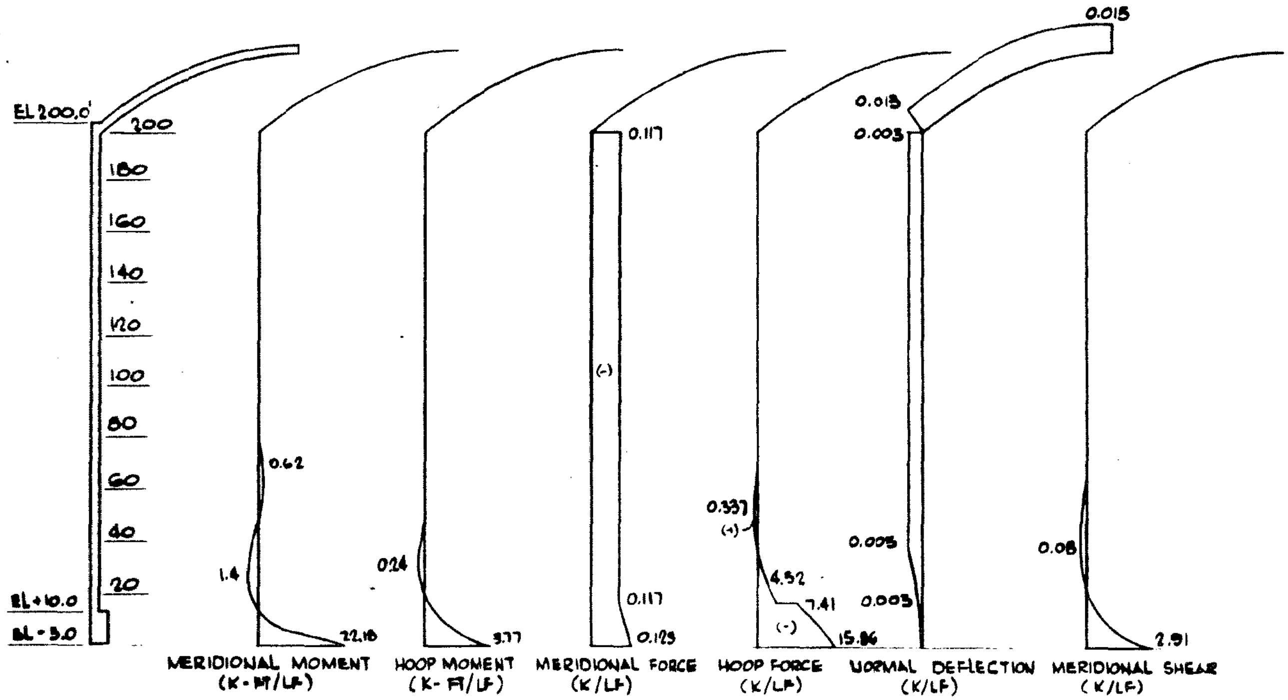
NOTE:

MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING
STRUCTURAL ANALYSIS
TORNADO WIND LOAD - 180°
FIGURE 3.8-38

UNIT TEMPERATURE - UNIFORM INCREASE



NOTE:

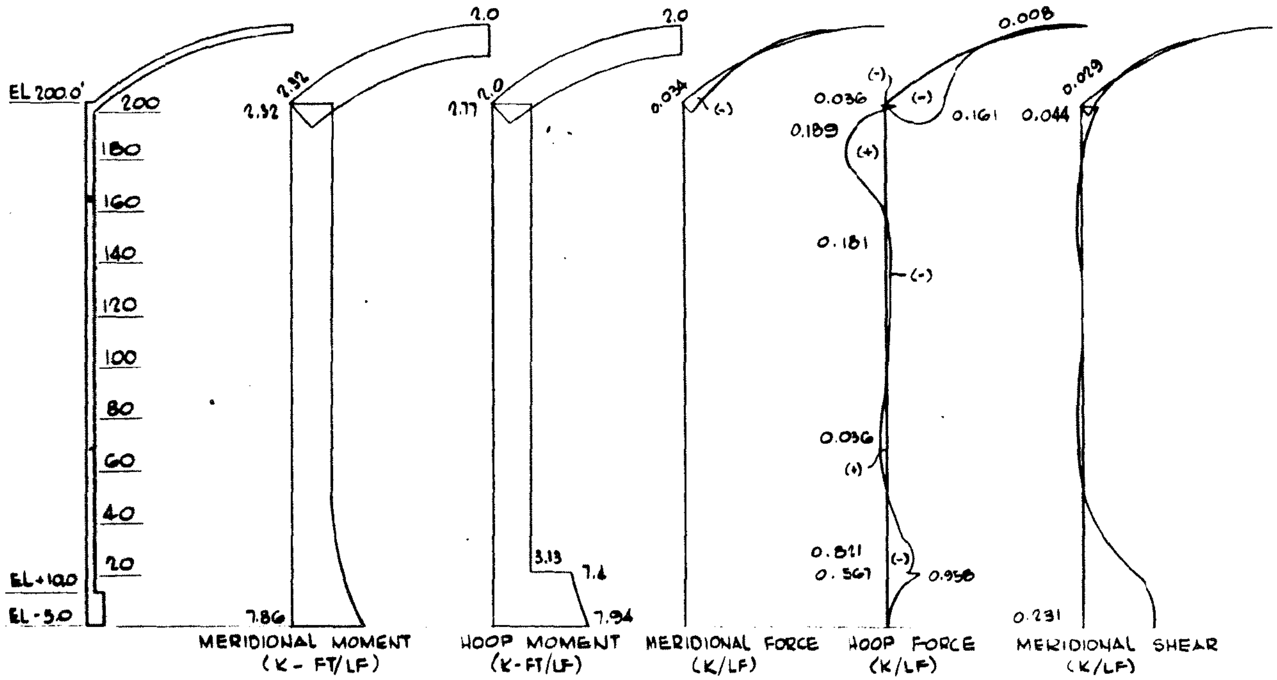
MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING STRUCTURAL ANALYSIS UNIT TEMPERATURE - UNIFORM INCREASE LOADING

FIGURE 3.8-39

UNIT TEMPERATURE - GRADIENT



NOTE:

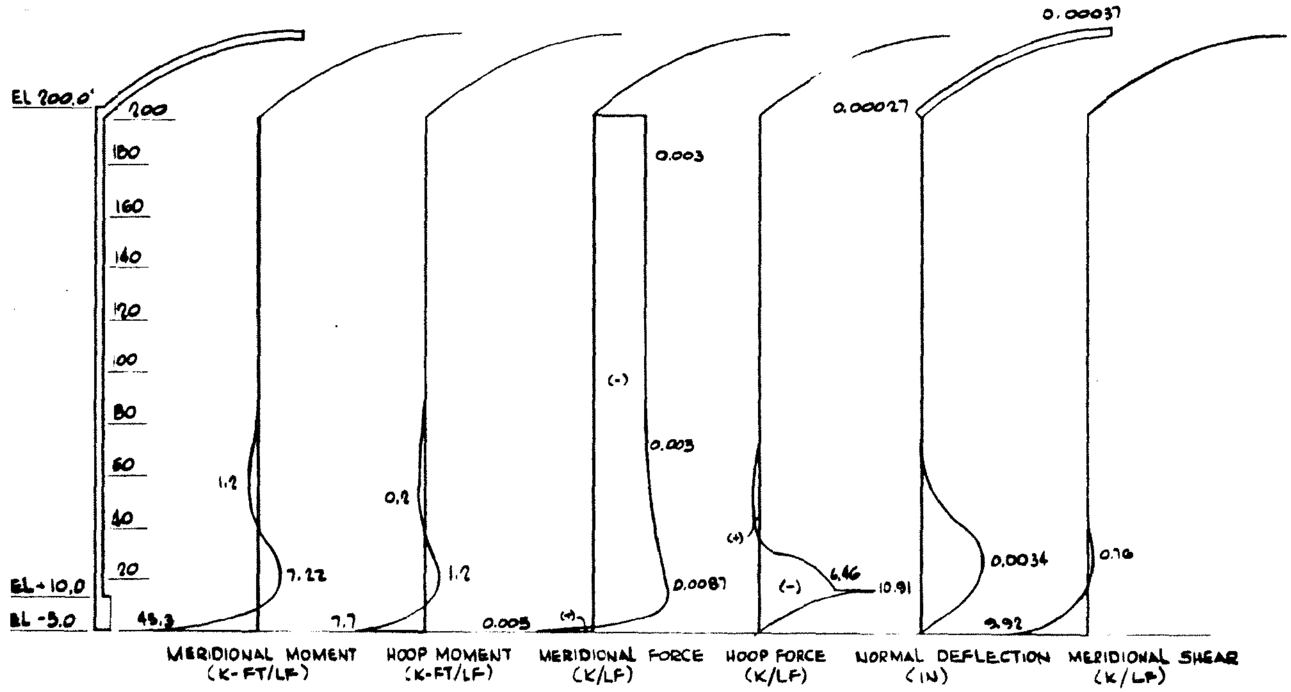
MOMENTS SHOWN ON TENSION SIDE
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING STRUCTURAL ANALYSIS
UNIT TEMPERATURE - GRADIENT LOADING

FIGURE 3.8-40

WATER TABLE @ EL +3'   GRADE @ EL +18'

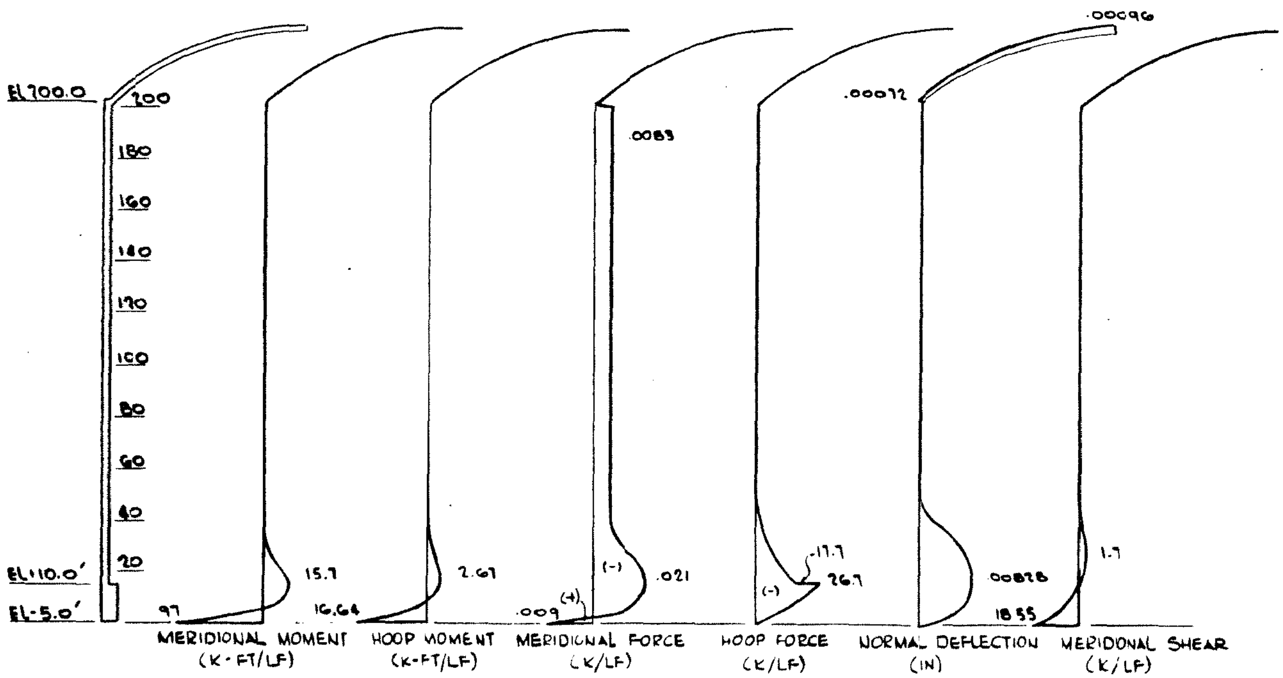


NOTE:
MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING
STRUCTURAL ANALYSIS
WATER TABLE AT EL +3' LOADING
FIGURE 3.8-41

WATER TABLE @ EL+21' GRADE @ EL+18'



NOTE:

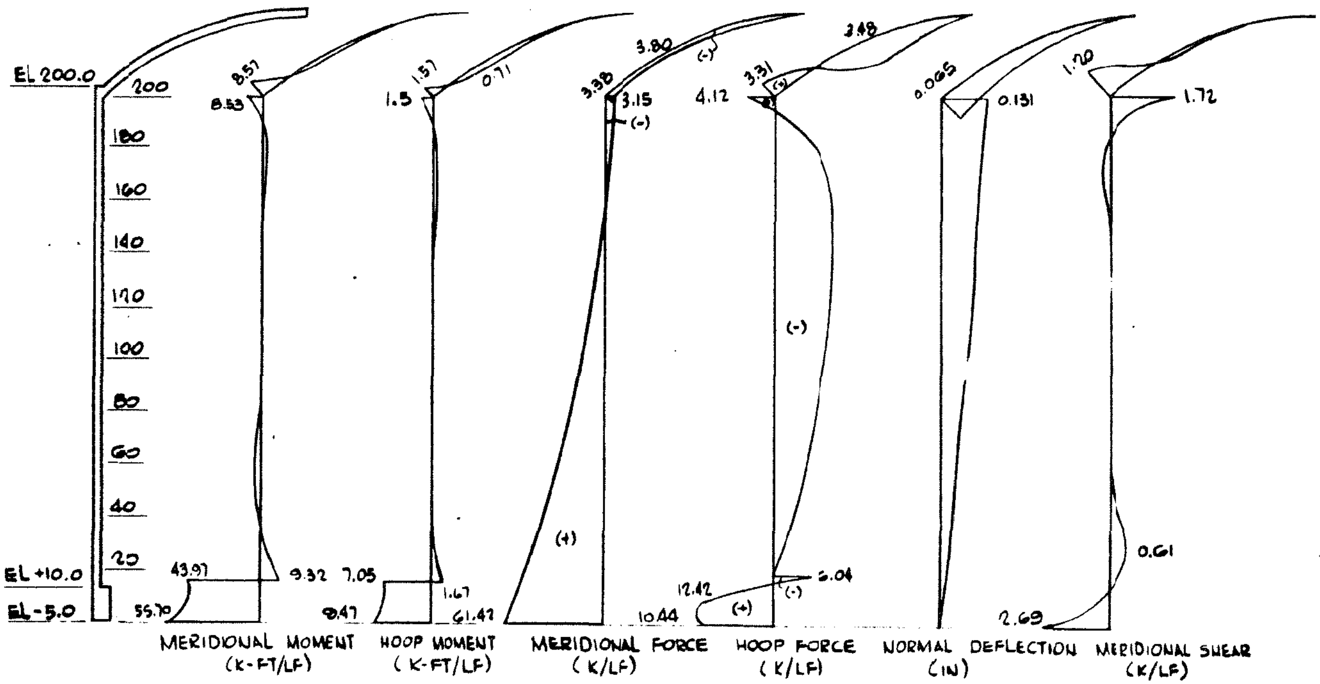
MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY
St. Lucie Plant

RESULTS OF SHIELD BUILDING STRUCTURAL ANALYSIS WATER TABLE AT EL +21' LOADING

FIGURE 3.8-42

1.0 EARTHQUAKE (0.1 g) θ=0°



NOTE:

MOMENTS SHOWN ON TENSION SIDE.
(+) INDICATES TENSION (-) INDICATES COMPRESSION

FLORIDA POWER & LIGHT COMPANY St. Lucie Plant
RESULTS OF SHIELD BUILDING STRUCTURAL ANALYSIS SEISMIC LOADING FIGURE 3.8-43

Security Related Information
Figure Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

TYPICAL CONCRETE
SUPPORT FOUNDATION

FIGURE 3.8-44

Refer to Drawing
8770-G-794

Florida Power & Light Company
St. Lucie Plant Unit 1

Reactor Building Equipment Supports
Figure 3.8-45

3.9 MECHANICAL SYSTEMS AND COMPONENTS

3.9.1 DYNAMIC SYSTEM ANALYSIS AND TESTING

3.9.1.1 Vibration Operational Testing

Safety related piping systems have been designed in accordance with ANSI B31.7 Code Class I, II, or III (see Table 3.2-2 for Code requirements).

In order to comply with the Code, each system has been designed to minimize dynamic effects but, due to the complexities of analyzing these effects, a preoperational test will also be implemented.

The preoperational test program for the Class I, II, and III piping systems will simulate actual operating modes to demonstrate that the appurtenances comprising these systems will meet functional design requirements.

Piping systems will be checked in three sequential series of tests and inspections. Construction acceptance, the first step, entails inspection of components for correct installation according to codes, specifications and drawings. During this phase, pipe and equipment supports will be checked for correct assembly and setting based on calculations. The cold locations of reactor coolant system components such as steam generators and reactor coolant pumps will be recorded.

During the second step of testing - plant heatup - the plant will be heated to normal operating temperatures. During the heatup, expansion data will be recorded and all systems will be observed periodically to verify proper expansion.

During the third step of testing - performance testing - systems will be operated and performance of the pumps, valves, controls, and auxiliary equipment checked. This phase of testing will include transient test, such as reactor coolant pump trips, reactor trip, and relief valve testing. During this phase of testing, the piping and piping restraints will be observed for vibration and expansion response. Automatic safety devices, control devices and other major equipment will be observed for indications of overstress, excess vibration, overheating and noise. Also, to verify the piping design for water hammer, each system test will include valve operation during transient system modes.

EPU implemented a Piping Vibration Monitoring Plan to ensure that any steady state flow-induced piping vibration and thermal expansion displacements on secondary systems piping were not detrimental to the plant, piping, pipe supports or connected equipment at pre-EPU/post modification, EPU power ascension and post-EPU conditions. The piping system vibration test plan excluded valve internal vibration, unstable check valve operation, active component operation, heat exchanger vibration and primary system piping vibration.

3.9.1.2 Seismic Testing and Analysis

Equipment specifications for seismic Class I mechanical component contained requirements for seismic testing or analysis. Initially, seismic forces in the horizontal and vertical direction ware determined by estimating the amplication of each floor acceleration due to the operational and design basis earthquake. These seismic loads were forwarded to the equipment manufacturer in the form of OBE loads (horizontal and vertical) and DBE loads (horizontal and vertical). The manufacturer was required to demonstrate that the equipment and equipment supports would not suffer loss of function under the maximum hypothetical loads.

In order to prove the seismic integrity of the Seismic Class I equipment, the manufacturer was required to perform one or more of the following:

a) Perform a vibration test over various frequencies associated with the DBE.

b) Submit calculations providing that the stress levels would not exceed the allowable stresses for the equipment and equipment supports.

c) Supply data experience for the equipment operating under similar loading conditions.

In addition, spectra curves (see Figures 3.7-12 through 3.7-24) indicating the floor response acceleration as a function of vibration period were forwarded to the manufacturer. The manufacturer was required to determine that the natural period of vibration for the equipment including its supports did not coincide with the critical period range of the equipment's floor spectra curve so that the actual loading including amplification of the supports was within the specified seismic acceleration values.

For seismic Class I mechanical components, Table 3.9-1 lists the specified g values, the accelerations for which the component was qualified and the method of qualification (test, analysis or operating data).

3.9.1.3 Reactor Internals - Core Support Barrel Evaluation & CSB/TS Vibration Analysis and Tests

Design analyses were performed on the reactor internals for normal operating conditions to demonstrate that the mechanical design bases defined in Section 4.2 were satisfied. These design calculations included appropriate vibration analyses of the component assemblies.

During the March, 1983 refueling outage, difficulties were encountered during core reload when a fuel assembly would not seat properly on the core support plate. Subsequent inspection determined there was debris of unknown origin on the plate. The fuel was unloaded and the core support barrel was removed to investigate the source of the debris.

A visual examination of the core support barrel/thermal shield assembly disclosed the thermal shield support system to be severely damaged. A number of thermal shield support pins were fractured and/or missing and damage to the core support barrel was visible. An evaluation of the thermal shield support system concluded that refurbishment was impractical. Analyses performed to evaluate operation of the plant without a thermal shield for its remaining design life indicated that replacement of the thermal shield was not necessary. Refer to Section 4.2.2.2 for details of thermal shield removal and re-analysis of the reactor vessel and internals.

Upon removal of the thermal shield from the core support barrel, a nondestructive examination of the core support barrel was conducted. Damage of varying degrees was in evidence at eight of the nine thermal shield support lug locations. Four lugs were separated from the core support barrel and through wall cracks were confirmed adjacent to some damaged lug areas.

Repair was made to stop propagation of the existing cracks in the core support barrel, to maintain bypass leakage at an acceptable level, and to assure core support barrel structural integrity remains.

The repair process for the core support barrel involved machining the damaged areas to reduce stress concentrations. Through wall cracks were arrested by crack arrestor holes appropriately sized for each crack; non-through-wall cracks were removed by machining away all material around the crack, and lug tear out areas were machined and patched as necessary. The crack arrestor holes and lug tear out areas were sealed by inserting expandable plugs and installing patches held in place by expandable plugs, respectively.

A re-analysis of the repaired core support barrel and the reactor internals without the thermal shield was performed. The component stresses under normal, upset and faulted conditions were evaluated and found to be within the limits of Section III, Subsection NG 1972, Draft Edition of the ASME Nuclear Components Code. The results of all analysis were submitted to the NRC (23), and were reviewed and concurrence obtained (24).

A re-analysis of the vibratory response of reactor internals to the hydraulic forcing functions was performed.

The normal operating loads, generated for use in the analysis, consist of the following categories of loads:

a. Static Hydraulic loads,
b. Pump Induced loads, and
c. Turbulence Induced loads.

These are discussed in detail in the following sections:

a) Static Hydraulic Loads

The static hydraulic loads acting on the portion of the core support barrel extending from the thermal shield lug elevation down to its bottom are given in Table 3.9-2a and Figure 3.9-6. Loads were calculated for the two sets of conditions given in Table 3.9-2b; the most adverse loads were chosen as input to the integrity analysis.

The axial hydraulic load in Table 3.9-2a is based on the operating conditions in Table 3.9-2b and calculated loss coefficients for the flow path segments between the inlet nozzles and the upper region. The maximum radial delta p across the core support barrel wall in Table 3.9-2a is also based on calculated loss coefficients.

The lateral loading distributions on the core support barrel given in Figure 3.9-6, are based on measured total pressures and flow kinetic heads in the downcomer region of a scaled flow model of the St. Lucie 1 reactor.

b) Pump-Induced Loads

Pump-induced acoustic loads acting on the core support barrel were calculated at an inlet temperature of 548°F at the following four pump characteristic frequencies:

1. rotor speed, 15 HZ

2. 2 x rotor speed, 30 HZ

3. blade passing frequency, 75 HZ

4. 2 x blade passing frequency, 150 HZ

The pump-induced loads on the core support barrel are determined using two hydrodynamic models:

1. The first model evaluates the propagation of pump-induced pressure pulsations in the cold leg water column from the pump discharge to the inlet nozzle on the vessel.

2. The second model evaluates the propagation of pump-induced pressure pulsations in the downcomer water column in the reactor vessel. The output from the first model is used to drive the second.

The wave equation for a compressible, inviscid fluid is set up and solved for each model. For the case of the downcomer, the series solution for the resulting 3-D wave equation was solved by means of a C-E computer code, DPVIB.

The output from the downcomer model consists of a description of the pressure distribution on the core support barrel wall, $P_o(R_{CSB}, \theta, Z)$. Typically, a pressure distribution is generated at each pump driving frequency for the case of a single operating pump with a nominal unit fluctuating inlet nozzle pressure. The resulting pressure distribution is described by the series:

$$P_o\left(R_{CSV}, \theta, Z\right) = \sum_m H_m \, COS \, m\theta$$

where:

H_m = $\sum_n \sum_s C_{nms} \left| J_m \left(\lambda_{nms}\Gamma\right) + \eta_{nms} \gamma_m \left(\lambda_{nms}\Gamma\right)\right| \cos \varepsilon_n Z$

m = Circumferential wave number

n = Axial wave number

s = Radial wave number

C_{nms} = Fourier coefficient

η_{nms} = Eigenvalue = $\left(W_{nms}^{2} / C_o^{2} - n^{2}\right)^{1/2}$

W_{nms} = Liquid natural frequency

C_o = Speed of sound in liquid

ε_n = Variable related to the axial waves n/L

r = Radius

$J_m \, Y_m$ = Bessel functions of first and second type

Z = Axial position

θ = Azimuthal position referenced to the zero degree position for the operating pump

L = Length of downcomer annulus

The pressure distribution P_o (R_{CSB}, θ, Z) based on the nominal unit psi inlet pressure, is scaled by the calculated inlet nozzle pressure that is output from the model for the cold leg.

$$P (R_{CSB}, \theta, Z) = P_{inlet} \times P_o (R_{CSB}, \theta, Z)$$

where:

P_{inlet} = calculated pump-induced pressure fluctuation at the vessel inlet nozzle; values are given in Table 3.9-2c

To obtain the overall pressure distribution P (R_{CSB}, θ, Z) on the core support barrel, for multiple pump operation, the pressure distribution for the single pump case was superimposed at the appropriate azimuthal positions for the particular operating pumps. To maximize pressure fluctuations on the downcomer, the phasing between the operating pumps was selected to produce the most adverse loading condition on the core support barrel.

c) Turbulence-Induced Loads

Hydraulic excitation of the core support barrel due to random turbulence was calculated from a power spectral density vs. frequency plot based on turbulent pressure fluctuation measurement in a scaled PWR model and coherence areas determined from laboratory and field test data inside a PWR.

In order to establish the structural integrity of the repaired core support barrel a comprehensive stress analysis was performed utilizing the methods and compared to the original criteria set forth in Section III, Subsection NG of the ASME Code: For Normal and Upset Conditions (Levels A and B), Figure NG-3221-1; For LOCA Conditions (Level D), Appendix F.

The following information is provided for historical information. The thermal shield removal has altered the results found herein. However, FPL in letter L-84-29 and the NRC in their March 14, 1984 letter have stipulated that the results contained herein are still conservative.

The flow induced vibration of the core support barrel, thermal shield system (CSB/TS), during normal operation, was characterized as a forced response to deterministic and random pressure fluctuations in the coolant. Methods were developed for predicting the response of components to the hydraulic forcing functions.

Emphasis was placed on analysis and design of those components which were particularly critical and susceptible to vibratory excitation, such as the thermal shield, Using a top supported, as opposed to a bottom supported, thermal shield design improves stability as it eliminates a free edge in the flow path. Increasing the number of upper supports and lower jackscrews, in the specific manner chosen, provides a much stiffer structure and the use of an all-welded shield eliminates local flexibilities and relative motion at bolted joints. Analytical studies show the thermal shield to be stable on its support system when exposed to the axial annular flow encountered during normal operation. The snubber design is based upon limiting the motion of the core support barrel under conditions of hydraulically induced vibrations. The snubbers are at the position of maximum amplitude for the fundamental lateral bending mode of the barrel, thereby restricting motion of the barrel at the most efficient position. The circumferential distribution of snubbers assures restraint regardless of the direction of response.

The random hydraulic forcing function was developed by analytical and experimental methods. An analytical expression was developed to define the turbulent pressure fluctuation for fully-developed flow. This expression was modified, based upon the result of scale model testing, to account for the fact that flow in the downcomer was not fully developed. Based upon test results, an expression was developed to define the spatial dependency of the turbulent pressure fluctuations. In addition, experimentally adjusted analytical expressions were developed to define the peak value of the pressure spectral density associated with the turbulence and the maximum area of coherence, in terms of the boundary layer displacement, across which the random pressure fluctuations are in phase.

The natural frequencies and mode shapes of the CSB/TS system were obtained using the axisymmetric shell finite element computer program, ASHSD (Reference 1). This computer program is capable of obtaining natural frequencies and mode shapes of complex axisymmetric shells: e.g., arbitrary meridional shape, verying thickness, branches, multi-materials and orthotropic material properties. To employ the ASHSD code, the core support barrel and thermal shield were modeled as a series of conical shell frustrums joined at their modal point circles. The length of each element, throughout the ASHSD model, was a fraction of the shell decay length. Since rapid changes in the stress pattern occur in regions of structural discontinuity, the nodal point circles were more closely spaced in such regions. The finite element model of the core support barrel, thermal shield system included representation of the core support barrel upper and lower flanges, sections of different wall thickness, and thermal shield support lugs and jackscrews.

Elements with orthotropic material properties were utilized to provide equivalent axisymmetric models of the structural stiffness and constraints to relative motion between the core support barrel and thermal shield provided by the thermal shield support lugs and jackscrews. Those modes which reflect the mass of the lower support structure, core shroud and fuel were simulated by the addition of concentrated masses at specific nodes in the core support barrel flange finite element model.

Applying Hamilton's variational principle to the conical shell elements and equation of motion was formulated for each degree of freedom of the system. An inverse iteration technique was utilized in the program to obtain solutions to the characteristic equation, which were based on a diagonilized form of a consistent mass matrix and stiffness matrices developed using the finite element method. Four degrees of freedom (radial displacement, circumferential displacement, vertical displacement, and meridional rotation) were taken into account in the analysis, giving rise to coupled mode shapes and corresponding frequencies. Evaluation of the reduction of these frequencies for the system immersed in coolant was made by means of the "virtual mass" method outlined in Reference 2.

The random response analysis considers the response of the CSB/TS system to the turbulent downcomer flow during steady-state operation. The random forcing function is assumed to be a wide-band stationary random process with a pressure spectral density equal to the peak value associated with the turbulence. The rms vibration level of the CSB/TS system was obtained based upon a damped, single degree of freedom analysis assuming the rms random pressure fluctuation to be spatially invariant. The snubber loads were derived from the random loads outlined above. Modeling the reactor vessel snubbers and core support barrel as a single degree of freedom spring-mass system, the number and magnitude of snubber, core barrel impacts was calculated based upon the response of the system to random excitation. The snubbers were designed, based upon this loading requirement, to meet the cyclic strength requirements specified in Section III of the ASME Boiler and Pressure Vessel Code.

The forced response of the reactor internals to deterministic loading was evaluated by classical analytical methods, using lumped mass and continuous elastic structural models. These calculated responses were used to verify the structural integrity of the reactor vessel internals to normal operating vibratory excitation. Components were analyzed to assure that there were no adverse affects from dominant excitation frequencies, such as pump rotational and blade passing frequencies, vortex shedding frequencies, and the natural frequencies of associated components.

The core support barrel is analyzed to provide assurance that this major structure does not exhibit excessive vibrations. It thereby keeps the entire reactor internals assembly at a low vibration level. Vibration analysis of the barrel, based on inlet flow impingement forces and turbulent flow, demonstrates that the anticipated rms response of the barrel is low.

Stresses in the most critical areas of the barrel are evaluated by superimposing thermal and static loadings on the dynamic response of the barrel. The calculated response of the fundamental mode of vibration of the core support barrel is used as an excitation in evaluating the response of the fuel assemblies to the core support plate and flow excitations. Vibration analysis of the assemblies demonstrates that the most likely modes of vibration do not coincide in frequency with available pump or mechanical excitation forces.

Experimental conformation of the adequacy of the reactor internals design is based upon the results of the prototype precritical vibration monitoring programs (PVMP) for Maine Yankee and Fort Calhoun. In accordance with AEC Regulatory Guide 1.20, prototype prediction, measurement and inspection programs were developed and performed for the Maine Yankee and Fort Calhoun reactor internals. Theoretical prediction analyses were performed for Maine Yankee (Reference 19) and Fort Calhoun (Reference 20) to estimate the amplitude, time and spatial dependency of the steady state and transient hydraulic and structural responses to be encountered during precritical testing. The PVMP for Maine Yankee has been completed successfully (Reference 21). The Fort Calhoun PVMP testing is complete. Reduction and analysis of the data is near completion.

The suitability of using PVMP data from Maine Yankee and Fort Calhoun programs as a composite prototype is based on the following:

a) C-E has provided predictive methodology and predicted limiting values of responses (acceptance criteria) for the Maine Yankee and Fort Calhoun prototype PVMPs (References 19 and 20, respectively).

b) The Maine Yankee and Fort Calhoun PVMP tests were completed successfully. The Maine Yankee results demonstrate the structural adequacy of the reactor internals for all normal, steady state and transient flow modes of reactor coolant pump operation. Fort Calhoun data is presently being analyzed.

c) Instrumentation to measure or derive forcing functions was included in the Maine Yankee and Fort Calhoun prototype reactor PVMPs in accordance with Regulatory Guide 1.20.

   The hydraulic forcing function prediction method has been verified by measurements on the Maine Yankee prototype PVMP (Reference 21).

d) The vibration test data, together with appropriate analyses, permit the assessment of those design differences which exist between Maine Yankee, Fort Calhoun and St. Lucie reactor internals, with respect to vibrational response characteristics.

Presented in Table 3.9-2 is a summary of the significant hydraulic and structural design parameters for each of the three reactor designs. St. Lucie is similar to Maine Yankee in structural size and component design, but differs from Maine Yankee in that it has top, rather than bottom mounted instrumentation, and is of two, rather than three loop design. St. Lucie has a minimum design flow rate identical to Maine Yankee, with other hydraulic parameters being similar (see Table 3.9-2). St. Lucie is similar to Fort Calhoun in that they are both of two loop design, with top mounted instrumentation. Fort Calhoun is smaller than St. Lucie and has a lower minimum design flow rate.

The effect of the aforementioned structural differences and normal design tolerances on the free vibration response of the reactor internals is, in general, small. The largest of the differences in natural frequencies occur for St. Lucie and Maine Yankee, in comparison with Fort Calhoun, which because of its smaller structure is stiffer, resulting in somewhat higher natural frequencies.

There is some significance to the differences in natural frequencies that do exist. Because of the circumferential positioning of the inlet and outlet nozzles, the two loop St. Lucie and Fort Calhoun systems would tend, with all four pumps operating, to an n = 2 circumferential mode, whereas Maine Yankee with three symmetrically spaced inlet nozzles would tend to an n = 3 mode configuration during normal operation. The dominant periodic excitation frequency in this frequency range is the pump rotational speed. Comparison of this excitation frequency to the appropriate natural frequencies, shows that St. Lucie can be expected to have the lowest dynamic, steady state magnification factor relative to excitation at the pump rotational speed.

In developing the analytical forcing functions for any number of reactor coolant pumps operating, it is assumed in the normal operating prediction analyses that it is possible to superpose the effect of one pump operating to obtain any multi-pump forcing function. During the Maine Yankee PVMP an investigation was made to ascertain the accuracy of the superposition procedure. Data obtained from single loop operation were combined to predict multiple loop pressures at various transducer locations. These values were compared with the actual measurements. The results indicate an average variation from perfect correlation of less that 25 percent. The majority of the predicted values exceeded the measured values, indicating conservatism in the estimates.

Based upon the above, it is seen that the "building block" approach is an acceptable procedure for developing multi-pump hydraulic forcing functions. Therefore, insofar as St. Lucie differs from Maine Yankee in number of loops, there will be no loss of accuracy in the prediction of the hydraulic forcing functions.

The Maine Yankee PVMP was conducted in accordance with Regulatory Guide 1.20 requirements for vibration monitoring of reactor internals of a prototype plant. The PVMP provided confirmation, based upon experimental evidence, that the hydraulic excitations and structural responses of the Maine Yankee reactor internals are within design estimates and are acceptable for all normal steady state and transient flow modes of reactor coolant pump operation. To accomplish this, the Maine Yankee PVMP provided measurements of the magnitude of core support barrel and thermal shield structural vibrations and hydraulic pressure fluctuations during various modes of reactor operation.

The Maine Yankee experimental program incorporated the use of internal and external accelerometers, pressure transducers, strain gauges, and scratch gauges, which permitted the recording at specific locations of time-dependent accelerations, pressures, strains, and maximum relative displacements. Measurements were made during functional testing on June 20-21, 1973 (hot pre-core), October 11, 1972 (cold pre-core), and October 17-18, 1972 (hot pre-core). The functional tests were of sufficient duration to assure satisfaction of the Regulatory Guide 1.20 requirement that critical components be subjected to a minimum of 10^7 cycles of vibration.

The Maine Yankee PVMP included a visual inspection program, with photographic documentation. Interface (and potential interface) surfaces of core support barrel, upper guide structure, core shroud, instrumentation nozzles, thermal shield supports, and reactor vessel were inspected to establish visually the condition of these components before and after functional testing.

The Maine Yankee PVMP also included an impact calibration test to correlate the response of those accelerometers externally mounted on the reactor pressure vessel with impulsive loads applied to the reactor vessel snubbers.

As part of the Maine Yankee vibration monitoring program, a theoretical analysis was performed to estimate the amplitude, time, and spatial dependency of the steady state and transient hydraulic forcing functions to be encountered during precritical testing. These forcing functions were used to determine analytically the dynamic response of the core support barrel, thermal shield system. The results of this analytical investigation were issued in a prediction report (Reference 19), which was submitted to the Commission on the Maine Yankee Docket, prior to initiation of the vibration monitoring program. Included in this prediction report were theoretical estimates of the forcing functions and associated structural responses, as well as definition of the PVMP design and test acceptance criteria, permissible deviations from the criteria and the bases upon which the criteria were established.

The final report of the Maine Yankee PVMP (Reference 21) summarizes the results of the experimental PVMP investigation, which includes functional and impact calibration testing, and the inspection program. In addition, this report summarizes a comparison of the experimental measurements with the analytically predicted responses and excitations, and presents an evaluation of the PVMP results with respect to design and test acceptance criteria.

From the Maine Yankee PVMP, it is concluded that;

a) Flow-induced vibrations of the Maine Yankee reactor internals are well within design allowable and are acceptable for all normal, steady state and transient flow modes of reactor coolant pump operation.

b) The theoretical prediction methods used provide conservative estimates of the total steady state and transient structural responses of the core support barrel, thermal shield system.

In accordance with Regulatory Guide 1.20, a test program of precritical vibration measurement and inspection of the Fort Calhoun reactor internals was developed for the purpose of demonstrating by test the acceptability of the reactor internals design for flow-induced vibrations under normal operating conditions. In the Fort Calhoun test program, a system of sensors consisting of pressure transducers, strain gauges and accelerometers was mounted on the core support barrel thermal shield system. A complementary analysis of the flow-induced vibration of the system under normal operating conditions was also performed. The methods of analysis used to predict the hydraulic forcing functions and the resulting vibratory response together with the results of that analysis, are presented in Reference 20.

Measurements were successfully made during functional testing on January 27, 1973 (cold pre-core), February 1, 1973 (hot pre-core) June 29, 1973 (cold post-core), August 5-6, 1973 (hot post-core). Analysis of the data is near completion.

Based upon the successful conclusion of the Maine Yankee PVMP and the current success of the Fort Calhoun PVMP, FP&L is proceeding to implement a PVMP for St. Lucie Unit 1 in accordance with the requirements of Regulatory Guide 1.20 as it relates to non prototype units.

In accordance with Regulatory Guide 1.20 requirements for a non prototype plant, the following testing and inspections will be performed for St. Lucie.

a) The reactor internals important to safety will be subjected during the preoperational functional testing program to all significant flow modes of normal reactor operation and under the same test conditions conducted on the Fort Calhoun, and Maine Yankee prototype designs.

   The test duration will be at least as long as it was for Fort Calhoun, and Maine Yankee.

b) Following completion of the preoperational functional tests, the reactor internals will be removed from the reactor vessel and visual and nondestructive examination of the reactor internals will be conducted. The areas examined will include:

   1) All major load-bearing elements of the reactor internals relied upon to retain the core structure in place.

   2) The lateral, vertical and torsional restraints provided within the vessel.

   3) Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals.

   4) Those other locations on the reactor internal components which were examined on the Fort Calhoun and Maine Yankee designs.

5) The interior of the reactor vessel for evidence of loose parts or foreign material.

A summary of the PVMP inspection and test program described above will be submitted to the Commission in report form.

The structural adequacy of the St. Lucie reactor internals will be demonstrated for normal steady state and transient flow modes of operation by the flow induced vibration analysis performed for it, in conjunction with the analytical and experimental results of the combined prototype precritical vibration monitoring programs performed for Maine Yankee and Fort Calhoun, and the non-prototype testing program for St. Lucie.

Due to the basic differences in the nature of the loading and response of the reactor internals under LOCA conditions as compared to flow induced vibrations during normal operation. The correlation of the measured results during preoperational vibration testing with the LOCA structural analyses and mathematical models is minimal.

During normal, operation tide vibratory response of the internals is characterized as a combination of random and steady forced response resulting from relatively low level random and deterministic pressure fluctuations in the coolant. The random pressure fluctuations arise from flow turbulence. The deterministic pressure fluctuations are pump related with dominant excitation frequencies at the pump rotational and blade passing frequencies and their harmonics.

During LOCA conditions the vibratory response is characterized as transient response resulting from rapidly varying high level pressure loading.

A feature common to both the LOCA and normal operating condition vibration analyses of the core support barrel system is the use of the modal superposition method requiring calculation of the CSB shell frequencies and mode shapes. However, preoperational vibration testing results to date have shown no significant or clearly identifiable response of the CSB system attributable to structural resonances. The results have shown that the significant responses occur at the deterministic pump related frequencies.

The ability of the LOCA mathematical models to represent vibrations is discussed and justified in Section 3.9.1.4 and reference 18.

The dynamic analysis methods used to evaluate LOCA effects on the reactor internals are described in reference 18. The seismic time history input used in the reference was that for the St. Lucie reactor internals. The method by which the seismic input excitation was obtained is discussed in Section 9.3.1 of references 18 and "Answers to Questions concerning CENPD-45 and CENPD-55." These are memorandums from C. L. Storrs, of Combustion Engineering to R. C. Young, of the AEC dated December 18, 1972. The summary of the analytical results for combined LOCA and seismic loads are presented in Tables 10.2, 10.3 and 10.4 of Reference 18.

3.9.1.3.1 Reactor Internals Vibration Monitoring (IVM) System (historical)

The reactor internals vibration monitor was a computer based system specifically designed to provide the operator with data on the status of the extent and character of core/core barrel motion.

The IVM system accepted isolated inputs from the excore linear channels and interpreted the stochastic responses in terms of internals motion by analyzing the attentuation noise.

The original requirement for the Reactor Internals Vibration Monitoring System stemmed from a design problem with the core barrel hold-down ring at several early vintage Combustion Engineering plants. For St. Lucie Unit 1 the hold-down ring was redesigned to provide additional force to hold the core barrel in place. In addition, the IVM System was installed and a surveillance program to monitor core barrel movement was implemented in the Technical Specifications. As originally anticipated in the Basis Section, the core barrel movement monitoring program was removed from the Technical Specifications (via Amendment #80) after several cycles of IVM data demonstrated that the redesign of the hold-down ring had been successful in preventing excessive core barrel movement. The IVM System was subsequently deleted since the function previously provided by this system was no longer needed.

3.9.1.4 Loss-of-Coolant Accident Blowdown Analysis

3.9.1.4.1 Analytical Method

ASHSD[l] is a structural finite-element computer code developed at the University of California, Berkeley, and supported in part by the National Science Foundation. It performs dynamic analyses of complex axisymmetric structures subjected to arbitrary dynamic loadings or base accelerations. The frequencies of free vibrations as calculated by ASHSD compare well to those calculated by the equations of Hermann-Mirshy and Flügge [3] [4], respectively. The authors also make comparisons with available experimental results,[5] of free vibrations of cylindrical shells. The resulting comparison is good. Comparison of the numerical solution,[6] of the dynamic response of a shell to suddenly applied loads and the finite-element (ASHSD) solution of the same problem are in good agreement. (1) The response of a shell to a moving axisymmetric pressure load was evaluated by ASHSD and analytically, (7) with the results being in good agreement.

SAMMSOR-DYNASOR[8] is a finite-element computer code developed at Texas A&M University and supported in part by a NASA grant from the Manned Spacecraft Center, Houston, Texas. This code has the capability of determining the non-linear dynamic response of axisymmetric shells subjected to arbitrary dynamic loads. Asymmetrical dynamic buckling can be investigated using this program. The program has been extensively tested, using problems the solutions to which have been reported by other researchers, in order to establish the validity of the codes. Among these are a shallow shell with axisymmetric loading as described in Reference 9. Identical results are obtained with those of Reference 10 for the analytical evaluation of blast loadings on a cylindrical shell. Calculations made by SAMMSOR-DYNASOR for the symmetric buckling of a shallow spherical cap is in good agreement with the analyses of References 11 and 12 and the experimental data of References 13 and 14.

SABOR-5 - DRASTIC[15] is a structural finite-element computer code developed at the Aeroelastic and Structures Research Laboratory, Department of Aeronautics at the Massachusetts Institute of Technology. The work was administered by the Air Force Systems Command with technical monitoring by the Aerospace Corp. SABOR 5 - DRASTIC is the end result of combining a finite-difference solution procedure and a finite-element program to permit predicting the transient response of complex shells of revolution which are subjected to arbitrary transient loadings. Comparisons with reliable independent analytical predictions (notably finite-difference transient response solutions submitted by AVCO) confirm the accuracy and reliability of the SABOR 5 - DRASTIC dynamic response predictions. An experiment and accompanying analysis were performed by the Aerospace Corp., [l6] to verify the ability of the code to account for a complex geometry shell of revolution subjected to transient asymmetric loads. Loads were applied by means of well defined explosive charges. Based upon the results of dynamic strain measurements made on the test structure, it is evident that the SABOR 5 - DRASTIC code is capable of solving complex dynamic shell structure problems successfully.

In developing the above finite-element computer codes, (i.e., ASHSD, SAMMSOR/DYNASOR, SABOR 5 - DRASTIC) the authors have independently verified their codes with respect to the results of other established structural programs, classical solutions and where possible to experimental data. The correlations demonstrate that the above programs are capable of solving complex dynamic shell structure problems successfully and that the finite-element method of modeling provides accurate representation of the structural phenomena. The SABOR 5 DRASTIC code, which has had extensive and successful analytical and experimental correlation[16] for transient (explosive) asymmetric loading, was used to analyze a core support barrel structure with short term loading. The results of this well-verified program are identical to those of the finite-element codes ASHSD and SAMMSOR /DYNASOR (which are used in the LOCA analysis) for the same core support barrel problem, demonstrating the ability of these programs to adequately represent and evaluate the effect of a transient load on an axisymmetric structure like the core support barrel.

3.9.1.4.2 Reactor Coolant System

The major components are designed to withstand the forces associated with postulated pipe ruptures, in combination with the forces associated with the Design Basis Earthquake and normal operating conditions.

The structural integrity of the reactor coolant system under LOCA loadings is confirmed using a static flexibility analysis in conjunction with a dynamic load factor.

The thrust loads developed by both guillotine and slot ruptures are applied to the reactor coolant system at those locations which, from previous analytical experience, impose the largest loading on components and/or supports.

The thrust force from guillotine ruptures is directed axially along the pipe. The thrust force from slot ruptures is directed radial to the ruptured pipe; the radial directions for slot ruptures at each location chosen are those which impose the largest loading on components and/or supports.

The magnitude of the thrust forces for guillotine and slot ruptures used in conjunction with the flexibility analysis is $P_{N.O.} \times A_p$,where $P_{N.O.}$ is the reactor coolant system pressure at normal operating conditions and A_p is the internal cross sectional area of the pipe.

Detailed thrust calculation from blowdown data generated by an approved LOCA code show the use of $P_{N.O.} \times A_p$ for the thrust force to be conservative. A dynamic load factor of 2 is applied to the thrust force to account for the dynamic response of the support system.

The mathematical model used in the flexibility analysis of the reactor coolant system is shown on Figure 3.9-1.

3.9.1.4.3 Reactor Internals

A dynamic analysis C-E proprietary topical report CENPD-42, has been performed to determine the structural response of the reactor vessel internals to the transient LOCA loading. The analysis determined the shell, beam and rigid body motions of the internals using established computerized structural response analyses. The finite-element computer code, ASHSD[1] was used to calculate the time-dependent beam and shell response of the core support barrel and thermal shield system to the transient LOCA loading. The finite-element computer code SAMMSOR-DYNASOR[4] was used to evaluate the core support barrel's potential for buckling when loaded by a net external radial pressure resulting from an outlet line break. The structural response of the reactor internals to vertical and transverse loads resulting from inlet and outlet breaks, was determined using the spring-mass computer code, SHOCK[2].

The time and space dependent pressure loads used in the above analysis were the result of a hydraulic blowdown analysis. The pressure fluctuations were determined for each node in the hydraulic model for inlet and outlet line breaks. The pressure time histories at these nodal locations were then decomposed into the Fourier harmonics which define the circumferential pressure distribution at the nodal elevations. Where the hydraulic model nodes did not correspond to those of the structural model, the hydraulic model pressure components were interpolated to provide the required loading information.

The finite-element computer code, ASHSD, was used to calculate the dynamic response of the core support barrel and thermal shield to transient LOCA loading resulting from an inlet break. To employ the ASHSD code, the core support barrel and thermal shield were modeled as a series of conical shell frustrums (elements) joined at their nodal point circles. Applying Hamilton's variational principle to the conical shell elements a damped equation of motion was formulated for each degree of freedom of the system. Four degrees of freedom;-- radial displacement, circumferential displacement, vertical displacement and meridional rotation---were taken into account in the analysis, giving rise to coupled modes. The differential equations of motions were solved numerically using a step integration procedure. To insure computational stability of the numerical solution, the integration time step was chosen such that it is small compared to the shortest period of the finite-element system. The model developed for the core support barrel, thermal shield system is shown in Figure 3.9-2. The length of each element, throughout the analytical model, was a fraction of the shell decay length. Since rapid changes in the stress pattern occur in regions of structural discontinuity, the nodal point circles were more closely spaced in such regions. The finite-element model of the core support barrel, thermal shield system included representation of the core support barrel upper and lower flanges, sections of different wall thickness, and thermal shield support lugs and jackscrews. Elements with orthotropic material properties were utilized to provide equivalent axisymmetric models of the structural stiffness and constraints to relative motion between the core support barrel and thermal shield provided by the thermal shield support lugs and jackscrews. Those modes which reflect the mass of the lower support structure, core shroud and fuel were simulated by the addition of concentrated masses at specific nodes in the core support barrel flange finite element model.

In performing the dynamic analysis of the core support barrel, thermal shield system, the transient load harmonics were applied in two successive phases to account for time-dependent boundary conditions at the snubbers. The first phase used those harmonics which excite the beam modes, whereas the second phase used those harmonics which excite the shell modes. During the first phase, the lower end of the core support barrel was unrestrained. Within a very few milliseconds, the clearance between the core support barrel and reactor vessel snubbers was closed and for the remainder of the LOCA transient, the core support barrel was restrained radially at the snubber level. Transient responses were computed throughout each loading phase.

The ASHSD code computed the nodal point displacement, resultant shell forces, shell stresses and maximum principle stresses as functions of time. The maximum principle stresses at the internal and external surfaces of the core support barrel and thermal shield were determined from the bending and membrane components during each phase of transient loading. Stress intensity levels calculated from the principle stresses were combined with normal operating and seismic induced stresses for comparison with design criteria specified in Section 4.2.2.

Accurate representation and analysis of the core support barrel and thermal shield shell structures was obtained through use of the finite element code ASHSD. Accurate representation of the remainder of the internals (i.e., fuel, core shroud, control element assemblies, upper guide structure and lower support structure) was obtained using the SHOCK code.[17]

The SHOCK code determines the response of structures which are represented as lumped mass systems and subjected to arbitrary loading functions. The code solves the differential equations of motion for each mass by a numerical step integration procedure. The lumped mass model can represent a vertically or laterally responding system subject to arbitrary loading functions and initial conditions. Options are available for describing steady state loads, preloads, input accelerations, linear and non-linear springs (including tension and compression only springs) gaps, and structural and viscous damping.

The reactor internals models are developed in terms of a spring mass system for both vertical and lateral directions; see Figures 3.9-3 and 3.9-4. For both models, the spring rates were generally evaluated using strength of material techniques. However, in complex areas such as at the core support barrel flanges and upper guide structure support flange, the stiffness was derived from finite element model analyses. The lumped-mass weights were generally based upon the mass distribution of the uniform support structures, but included at appropriate nodes, local masses such as snubber blocks, fuel assembly end fittings, thermal shield lugs, etc. The net result was a lumped-mass system having the same distribution of mass as the actual structure. To simulate the effect associated with the internals oscillating laterally in the water filled vessel, a distributed virtual mass was calculated based upon the procedure outlined in Reference 3 (which includes the annulus effect) and was added to the structural lumped-mass system, to provide an analytical model with a dynamic response quantitatively similar to the actual internals. In the case of the vertical model, the hydraulic effect is notably one of reducing the effective weight of the reactor internals and this effect was included in the structural lumped-mass system.

The SHOCK code provided excellent facility for modeling clearances, preloads and component interfaces. In the lateral model, the core support barrel, reactor vessel snubber clearance was simulated by a non-linear spring which accounted for the increased resistance to core support barrel motion when snubbing occurred. In the vertical model, non-linear springs in the form of compression only springs, were used extensively to simulate preload and interface conditions, such as exist between the upper guide structure support plate and core support barrel upper flange; at the fuel hold-down spring; at the fuel, core support plate interface and at the core shroud, core support plate interface. Tension springs were used to simulate the effect of the core shroud tie rods.

In both the vertical and lateral SHOCK models damping was varied throughout the system to simulate structural and hydraulic frictional effects within the reactor internals. The effect of hydraulic drag in the vertical model was simulated by a force-time history applied to the fuel assembly lower end-fitting. Vertical loads were used directly from the detailed hydraulic analysis, whereas lateral loads were obtained by integrating those harmonics which excite the beam modes to obtain the net lateral load on the core support barrel, thermal shield system.

The SHOCK code calculated the vertical and lateral response of the system in terms of displacements, velocities and accelerations and internal force, moments and shears as related to each model. These quantities were sufficient to permit calculation of membrane and where appropriate bending stresses for comparison with criteria specified in Section 4.2.2.

The finite-element code SAMMSOR-DYNASOR was used to determine the dynamic response of the core support barrel, with initially imperfect geometry, to a net external radial pressure resulting from an outlet line break. The above analysis has the capability of determining the non-linear dynamic response of axisymetric shells with initial imperfections subjected to arbitrarily varying load configurations.

Since SAMMSOR-DYNASOR is a finite-element program, a model was developed, Figure 3.9-5, of the core support barrel using axisymmetric finite-elements similar to those used for the ASHSD analysis. As was for the ASHSD model, the SAMMSOR-DYNASOR finite-element lengths were considerably less than the decay length of the core support barrel. The boundary condition at the core support barrel flange was considered fixed, whereas at the core support barrel lower flange radial displacements were restrained. These boundary conditions represented the restraint due to the expansion compensating ring and pressure vessel head at the top and the snubbers and lower support structure at the bottom. For conservatism, the stiffening effects of the fuel alignment plate, core shroud and core support plate were neglected.

Since the basic phenomenon in buckling is non-linear instability, the initial deviation of the structure from a perfect geometry greatly effects its response. The initial imperfection was applied to the core support barrel by means of a pseudo-load so developed to provide the maximum imperfection over each of the desired number of circumferential harmonics. The actual transient loading in terms of its harmonics was applied to the initially "imperfect" geometry core support barrel and the response obtained for each of the imperfection harmonics for the combined loading harmonics.

The input to the reactor internals pipe break analysis is the reactor coolant system LOCA analysis and blow-down loads, which were not revised for EPU conditions. Therefore, the reactor internals are not re-analyzed for LOCA events, as the pre-EPU analysis remains applicable for EPU.

3.9.1.5 Stress Analysis Methods

The system or subsystem analysis used to establish, or confirm, loads specified for the design of reactor coolant system components and supports was performed on an elastic static basis using a dynamic load factor of 2 as described in Section 3.9.1.4. The analysis on the reactor vessel internals is based on the criteria discussed in Section 4.2.2. The analysis used for the reactor coolant system design is discussed in Section 5.2.1. In the design approach, based upon the results of analysis obtained for the preliminary design, the system was modified by changing support stiffness and adding restraints as required to control resonance conditions and to maintain the response of the system appropriately within the elastic range. Elastic stress analysis methods were also used in the design calculations to evaluate the effects of the loads on the components and supports.

Dynamic analysis as described in Sections 3.7.3 and 3.9.1 is performed to verify that the stresses are within limits specified by the applicable code requirements. Allowable stresses for Code Class 1, 2 and 3 components are given in Table 3.9-3.

C-E, "Final Report-Dynamic Seismic Analysis Of The Reactor Coolant System Components for Florida Power And Light Company, St. Lucie Plant Unit 1," dated October 1972, describes the modeling methods, computer codes used and the results of the analysis which show that all calculated loads produced by seismic excitation are less than specified for component design. These results include typical diagrams of models used in the analysis and the reactor coolant unbroken loop piping under simultaneous LOCA and DBE loads.

3.9.1.6 Stress Analysis Results For Repaired Core Support Barrel

After repair of the Core Support Barrel and removal of the Thermal Shield in 1983, a stress analysis was performed to verify acceptability of repairs.

The analysis was performed for the region of the core support barrel at the thermal shield lug elevation. The conservative assumption was made that at each of the lug regions the maximum length of lateral crack was circumferential and in the same horizontal plane as the cracks in the other lugs. The point of maximum stress in the region was then established by determining the axis in the plane about which the moment of inertia of the cylindrical section in combination with the load resulted in the maximum stress. The fatigue analysis was performed utilizing the stress concentration factors resulting from the crack arrestor hole size analysis. The design fatigue curves used in the analysis are the more conservative fatigue curves published in the Winter 1982 Addenda to Section III, Appendix I, Figures I-9.2.1 and I-9.2.2.

In addition to the Code Analysis, a confirmatory stress analysis of the core support barrel was performed using sophisticated finite element techniques. Overall effects and local effects of cracks in the core support barrel were evaluated by comparing stress distributions to those of an uncracked barrel. The conclusion of the confirmatory analysis was that the analysis considering the horizontal crack length in the same horizontal plane was conservative.

A summary of the Code Analysis results is shown in Table 3.9-3b.

An evaluation of the cracks in the core support barrel on the basis of fracture mechanics considerations was performed. After discussion with consultants on fracture mechanics it was concluded that insufficient data for the barrel material in a pressurized water reactor environment for service in excess of 10^{11} cycles was available. Because of the lack of materials data and the length of cracks in the core support barrel extremely conservative assumptions would have had to be made. The decision was made to use crack arrestor holes sized to reduce stress concentrations to magnitudes compatible with the ASME code fatigue limitations.

The stress concentration factors for a crack with a crack arrestor hole at each end were calculated using available theoretical solutions of stress distributions in plates with openings. (23) The adequacy of the solutions was verified through comparisons with finite element analyses of typical crack geometries and loading conditions.

The "equivalent ellipse" concept is useful in calculating stress concentration factors for a crack with crack arrestor holes at each end. For an elliptical hole in an infinite plate in tension, the stress concentration factor, K_t, is given by:

$$K_t = 1 + 2\sqrt{\frac{b}{2r}}$$

Where: b = major length of elliptical hole
r = minimum radius of elliptical hole

Core support barrel temperature is also an important consideration in the integrity analysis since temperatures give rise to thermal stresses that must be considered in the core support barrel analysis.

A three-dimensional ANSYS finite element model was used to determine temperatures for evaluation of core support barrel integrity. The thermal analysis was performed with appropriate heat generation rates, bulk fluid temperatures and heat transfer film coefficients to obtain the temperature distribution in the core support barrel.

The energy deposition heating rate was based on a reactor operating power of 3020 MWt with spatial core power distributions designed to produce conservative results for components in radial locations outboard of the reactor core, such as the core support barrel, with a 15% analytical uncertainty allowance.

The core support barrel heating rate calculations were completed using the TORT (three-dimensional) discrete ordinates codes from the DOORS 3.2 Code Package. This suite of codes has been used to support numerous pressure vessel fluence evaluations and is accepted by the Nuclear Regulatory Commission (NRC) for deterministic particle transport calculations. The transport cross-sections used in the calculations were taken from the BUGLE-96 coupled neutron/photon cross-section library that was generated specifically for Light Water Reactor (LWR) applications.

Stresses based on the results of the thermal analysis for EPU conditions have been used in the core support barrel fatigue analysis and provided acceptable results.

3.9.1.7 Program for Monitoring of Thinning of Pipe Walls of High Energy Carbon Steel Piping

In response to the feedwater pipe rupture event at the Surry Plant and the issuance of IE Notice 86- 106 and IE Bulletin 87-01, a program for monitoring pipe wall thinning in carbon steel piping due to erosion/corrosion has been developed. Generally, piping wall thicknesses are monitored to ensure that code requirements for wall thickness are satisfied. The program includes all moderate and high energy piping systems, both nuclear safety related and non-nuclear safety related.

Inspection locations are established in accordance with accepted industry methods such as those provided by EPRI for single and two phase systems. Within specific piping systems, locations for inspections are selected based upon such factors as fluid velocity, temperature, moisture content (for steam), chemistry, material composition, and piping geometry. Areas which are subjected to flow disturbances such as elbows, branch connections and piping and fittings downstream of control valves or flow orifices are preferred locations for inspections.

The program is designed to first inspect the most likely points for erosion/corrosion and to collect "baseline" data on other locations, with program expansion required if wall thinning in any location was more severe than anticipated. Frequency of inspection is based upon the rate of erosion /corrosion. Each operating cycle, inspection data is reviewed to determine which locations, based upon measured maximum erosion/corrosion rates, may be approaching code minimum wall thickness values.

The method of examination is selected based upon the ability to accurately provide a profile of wall thickness readings over the entire area of the piping or fitting expected to experience significant erosion/corrosion. In general, ultrasonic devices have been used for this purpose.

Decisions to take corrective action for piping and fittings which have suffered erosion/corrosion damage are based upon the ability of the piping or fitting to satisfy code minimum wall thickness requirements during the subsequent operating cycle. If the lowest wall thickness reading in a piping section less the erosion/corrosion expected during the subsequent operating cycle is less than the minimum value required by the applicable code, the piping section must be repaired or replaced.

The NRC, in Reference 29 confirms that the implemented erosion/corrosion program meets the requirements of Generic Letter 89-08.

3.9.2 ASME CODE CLASS II AND III COMPONENTS

3.9.2.1 Design Conditions

The design pressure, temperature and other conditions that were considered in the design of each system containing Code Class 2 or 3 mechanical components are listed in Table 3.9-4.

3.9.2.2 Design Loading Combinations

The design loading combinations considered in the component design are: normal (operating design) pressure, temperature and thrust loads combined with seismic, hurricane or tornado loads. Seismic loads and hurricane and tornado loads are not assumed to act concurrently. The design loading conditions are categorized as design, normal, upset, emergency, and faulted. The stress limits associated with each of the design loadings categories Code Class 2 or 3 components are given in Table 3.9-3A, and for piping in Table 3.9-3.

The forces and moments acting on any component in the piping system are supplied to the manufacturer so that it can be insured that the component will function under the applied loads

Loads resulting from transients appropriate to specified plant operating conditions have been considered and accommodated by design. These conditions have been analyzed in accordance with applicable code requirements as an independent case. The transient operating conditions accounted for in the design of the reactor coolant pressure boundary (NSS vendor's scope) is provided in Section 5.2.1.2. Cyclic loading considerations for equipment outside the NSS vendor's scope is discussed below.

The ASME code does not require cyclic analyses for Class 2 and 3 components. Equipment specifications for pumps specify "maximum" moments and forces at the pump nozzles. These maximum moments and forces envelop operating transient loading conditions appropriate for the component. (See Table 3.9-3A footnote 2). For Class 2 and 3 piping the dynamic conditions resulting from fast valve closure and relief valve operation are analyzed as shown in loading combination 3 of Table 3.9-3. These dynamic conditions envelop the operating transients.

For Class I piping and fitting assemblies, fatigue analysis has been performed to ensure the usage factor is adequate for the 60-year design life. The applicable transients have been assigned operating condition categories, normal (N), upset (U), test(T), emergency (E), or faulted (F). Cyclic loading combinations considered for Class I piping and assemblies include:

Lines Associated with Pressurizer (For Pressurizer transients see Ref. 31)

Operating Condition Category	Condition	Lifetime Occurrences
N	Pressurizer Heatup	500
N	Pressurizer Cooldown	500
N	Plant Loading - 5%/min	15,000
N	Plant Unloading - 5%/min	15,000
N	10% Step Load Increase	2,000
N	10% Step Load Decrease	2,000
U	Reactor Trip	400
T	Hydrostatic Test, 3125 psia, 100 - 400 F	10
T	Leak Test 2250 psia, 100 - 400 F	200
N	Normal Pressure Variations ±100 psi, ±7 F	10^6
U	Loss of Primary System Flow	40
E	Loss of Secondary Pressure	5
U	Loss of Turbine-Generator Load	40

Pressurizer relief valve piping will be subject to the above conditions at the pressurizer nozzle. In addition, the piping will be subject to 40 cycles of steam flow per a specified transient which is categorized as an "Upset" operating condition.

With the temperature of the pressurizer and spray nozzle of 440 F and the pressure at 250 psig, cold spray water at 40 F may be introduced at the rate of 133 gpm. This transient will occur a maximum of 500 times during the lifetime of the plant.

During plant cooldown auxiliary spray piping I-2RC-149 and I-2-CH-146 will be subjected to 500 lifetime occurrences of the following temperature transient:

a) Cooldown from 532 F to 300 F at 75 F/hr - No flow

b) Cooldown from 300 F to 120 F in 1.5 seconds - 0 to 44 gpm

c) Depressurize and collapse bubble 120 F,> 1 hr - 44 gpm

This operating condition is categorized as "normal".

In addition to the above, the auxiliary spray piping will also be subjected to 16 occurrences of cooldown with system loss of flow as follows:

a) Gradual heatup from 532 F to 650 F in 8 hrs - no flow

b) Cooldown from 650 F to 120 F in 1.5 seconds - 0 to 44 gpm

c) Depressurize to 1600 psia 120 F,> 1 hr - 44 gpm

d) Gradual heatup to 604 F - No flow

e) Depressurize from 604 F to 120 F in 1.5 seconds - 0 to 44 gpm

This condition is categorized as "upset".

In addition to the transients described above, the surge nozzle will also be subjected to the following transients as a result of thermal stratification in the surge line:

Condition	Lifetime Occurrences
Heatup, ΔT = 340°F, Low Pressure	75
Heatup, ΔT = 340°F, High Pressure	75
Heatup, ΔT = 250°F, Low Pressure	375
Heatup, ΔT = 250°F, High Pressure	375
Heatup, ΔT = 200°F, Low Pressure	400
Heatup, ΔT = 200°F, High Pressure	400
Heatup, ΔT = 150°F, Low Pressure	500
Heatup, ΔT = 150°F, High Pressure	500
Heatup, ΔT = 90°F, Hot Standby	87710
Cooldown, ΔT = 340°F, Low Pressure	75
Cooldown, ΔT = 340°F, High Pressure	75
Cooldown, ΔT = 250°F, Low Pressure	375
Cooldown, ΔT = 250°F, High Pressure	375
Cooldown, ΔT = 200°F, Low Pressure	400
Cooldown, ΔT = 200°F, High Pressure	400
Cooldown, ΔT = 150°F, Low Pressure	500
Cooldown, ΔT = 150°F, High Pressure	500
Cooldown, ΔT = 90°F, Hot Standby	87710

Surge line thermal stratification was evaluated by the CE Owners Group (CEOG). The data above comes from that analysis. ΔT is defined as being Pressurizer Temperature minus RCS Hot Leg Temperature. Low pressure is defined as 410 psia and high pressure is defined as 2,250 psia. Starting temperatures for the low pressure transients are 440°F for the pressurizer and 100F for the RCS hot leg (ΔT = 340°F). Starting temperatures for the high pressure transients are 653°F for the pressurizer and 313°F for the RCS hot leg (ΔT = 340°F). Changes in surge line temperature were evaluated to assess the impact of thermal stratification on the surge line to demonstrate that it meets the applicable design codes and other Final Safety Analysis Report and regulatory commitments for the design life of the plant.

Lines Associated with Steam Generator (For Steam Generator transients see Ref. 31)

Operating Condition Category	Condition		Lifetime Occurrences
N	a –	Plant Heatup 100 F/hr	500
N	b –	Plant Cooldown 100 F/hr	500
N	c –	Plant Loading, 5% /min	15,000
N	d –	Plant Unloading, 5% /min	15,000
N	e –	0% Step Load Increase	2,000
N	f –	10% Step Load Decrease	2,000
U	g –	Reactor Trip	400
T	h –	Primary side hydrostatic test, 3125 psia, 100 - 400 F	10
T	i –	Secondary side hydrostatic test, 1250 psia	10
T	j –	Primary side leak test, 2250 psia, 100 - 400 F	200
T	k –	Secondary side leak test, 1000 psia	200
N	l –	Cold Feed Following Hot Standby	15,000
N	m –	Normal Plant Variations, ± 6 F primary, ± 40 psi secondary	10^6
N	n –	Primary Coolant Pump Starting and Stopping	4,000
U	o –	Loss of Primary Flow	40
U	p –	Loss of Turbine Generator Load	40
E	q –	Loss of Secondary Pressure	5
E	r –	Loss of Feed Flow	8

Reactor Vessel Head Vent Line (For Reactor Vessel Head transients see Ref. 31)

Operating Condition Category	Condition		Lifetime Occurrences
N	a –	Heatup, 100 F/hr	500
N	b –	Cooldown, 100 F/hr	500
N	c –	Loading, 5%/min	15,000
N	d –	Unloading, 5%/min	15,000
N	e –	Step Load Increase, 10%	2,000
N	f –	Step Load Decrease, 10%	2,000
U	g –	Reactor Trip, Loss of Load	400
T	h –	Hydrostatic Test, 3125 psia, 100 - 400 F	10
T	i –	Leak Test, 2250 psia, 100 - 400 F	200
N	j –	Normal Plant Variations, ±100 psi, ±6 F	10^6
U	k –	Loss of Primary System Flow	40
U	l –	Abnormal Loss of Load	40
E	m –	Loss of Secondary Pressure	5

Reactor Coolant Loop Tributary Piping (For Reactor Coolant Piping transients see Ref. 31)

a. Miscellaneous Pressure, Drain and Sample Lines

Operating Condition Category		Plant Conditions	Occurrences
N	a –	Heatup, 100 F/hr	500
N	b –	Cooldown, 100 F/hr	500
N	c –	Loading, 5%/min	15,000
N	d –	Unloading, 5%/min	15,000
N	e –	Step Load Increase, 10%	2,000
N	f –	Step Load Decrease, 10%	2,000
U	g –	Reactor Trip	400
T	h –	Hydrostatic Test, 3125 psia, 100 - 400 F	10
T	i –	Leak Test, 2250 psia, 100 - 400 F	200
N	j –	Normal Plant Variations ±100 psi, ±6 F	10^6
U	k –	Loss of Reactor Coolant System Flow	40
U	l –	Loss of Turbine-Generator Load	40
E	m –	Loss of Secondary Pressure	5

b. Charging Lines

Continuous injection of 395 F water into the reactor coolant cold leg piping at 551 F and 2250 psia at a rate of 22 gpm per nozzle. This shall be categorized as "normal" operating condition.

Also under normal operating conditions, 500 cycles of on and off charging flow at 40 gpm and 40 F with the loop at 551 F. This shall be categorized as "normal" operating condition.

Injection of charging water into the reactor coolant cold leg piping during the following operations:

Operating Condition Category		Operating	Occurrences
N	a –	Plant Heatup Transient 100 F/hr	500
N	b –	Plant Cooldown Transient 100 F/hr	500
N	c –	Plant Loading Transient 5%/min	15,000
N	d –	Plant Unloading Transient 5%/min	15,000
N	e –	Plant Step Load Increase ± 10% P F	2,000
N	f –	Plant Step Load Decrease ± 10% F P	2,000
U	g –	Plant Reactor Trip Transient	400
U	h –	Plant Loss of Flow Transient	40
U	i –	Plant Loss of Load Transient	40
E	j –	Plant Loss of Secondary Pressure Transient	5

Operating Condition Category	Operating			Occurrences
N	k	–	Purification	1,000
N	l	–	Low Volume Control & Makeup	2,000
N	m	–	Boric Acid Dilution	8,000
U	n	–	Loss of Charging Flow	200
U	o	–	Loss of Letdown	50
U	p	–	Regenerative HX Isolation Long Term	80
U	q	–	Regenerative HX Isolation Short Term	40

C. Safety Injection Supply Lines

The safety injection lines I-6-SI-110, 111, 112, 113, I-12-RC-151, 152, 153, 154 from valves V3114, V3124, V3134 and V3144 to the appropriate cold leg nozzle are each as a normal operating condition subject to five hundred (500) injections of 200°F water into the primary coolant cold leg piping initially at 300°F with a system pressure of 300 psia or less at a rate of 1500 gpm per nozzle. (from low pressure safety injection pumps)

As an "Emergency" operating condition, the safety injection lines:

I-3/4-SI-114, 115, 116, 117	Instrument Lines
I-3-SI-137, 138, 139, 140	HP & Aux HP Lines
I-2-SI-126, 143, 145, 147	HP & Aux HP Lines
I-6-SI-110, 111, 112, 113 I-12-RC-151, 152, 153, 154	Safety Injection Lines
I-1-SI-227 thru 246	Vent & Drains

covering the high pressure and auxiliary high pressure headers from valves V3113, V3123, V3133 and V3143 through the appropriate cold leg nozzle may be subject to five (5) injections of 120°F water into the reactor coolant cold leg piping initially at 551°F with a system pressure of 1240 psia or less at a rate of 225 gpm per nozzle for 90 seconds followed by 40°F water from High Pressure Safety Injection Pumps.

The safety injection tank discharge lines I-12-SI-148, 149, 150, 151, I-6-SI-110, 111, 112, 113 and I-12-RC-151, 152, 153, 154 from the check valves below the tanks to the cold leg nozzles will, as a faulted condition, be subject to one (1) injection of 100°F water into the primary coolant cold leg piping initially at 551°F with a system pressure in excess of 230 psig or less at a rate of 19,000 gpm per nozzle. Flow decreases linearly over 25 seconds to 2000 gpm per nozzle (from safety injection tanks).

Following the above flow at 40°F from the low pressure safety injection headers through the safety injection lines to the cold leg nozzles will be maintained at 2000 gpm per line (from low pressure safety injection pumps)until equilibrium is reached. This is a "faulted" operating condition.

d. Safety Injection Return Lines

The safety injection return lines (I-1-SI-118, I-1-SI-120, I-1-SI-123 and I-1-SI-125) are subject to 2000 occurrences of a step change from 130°F and 1100 psia to 120°F and 200 psia. This transient occurs upon opening the return line pneumatic valves to relieve the pressure accumulated between the safety injection check valves (V3113, V3114 and V3217 typical). The flow rate varies from 0 to 40 gpm during those step changes. This transient occurs periodically during the operation of the plant.

e. Shutdown Cooling Suction Lines

The shutdown cooling suction lines, I-12- RC-147 and 162, I-10-SI-127 and 130, as a normal operating condition, be subject to 500 occurrences of shutdown cooling with a flow of 3000 gpm, an initial temperature of 350°F max and pressure and temperature varying as appropriate for cooldown beyond 350 °F.

f. Letdown Line

Five hundred (500) heat-up cycles with a flow of 80 gpm and temperature increasing at 100°F/hr from 70°F to 550°F and pressure increasing from atmospheric to 2250 psia over this period. This condition should be considered as a "normal" operating condition.

Five hundred (500) cooldown cycles of flow at 29 gpm and temperature decreasing from 550°F to 140°F at a rate of 100°F/hr and pressure decreasing from 2250 psia to atmospheric. This condition should be considered as a "normal' operating condition.

Operating Condition Category	Plant Conditions	Occurrences
N	a – Heatup, 100°F/hr	500
N	b – Cooldown, 100°F/hr	500
N	c – Loading, 5%/min.	15,000
N	d – Unloading, 5%/min.	15,000
N	e – Step Load Increase, +10%	2,000
N	f – Step Load Decrease, -10%	2,000
U	g – Reactor Trip	400
U	h – Loss of Reactor Coolant System Flow	40
U	i – Loss of Turbine-Generator Load	40
E	j – Loss of Secondary Pressure	5
N	k – Purification	1,000
N	l – Low Volume Control & Makeup	2,000
N	m – Boric Acid Dilution	8,000
U	n – Loss of Charging Flow	200
U	o – Loss of Letdown	50
U	p – Regenerative Hx Isolation Long-term	150
U	q – Regenerative Hx Isolation Short-term	40

Lines Associated with Reactor Coolant Pumps (For Reactor Coolant Pump transients see Ref. 31)

Plant Conditions		Occurrences
a.	Heat Up, 100°F/hr	500
b.	Cooldown, 100°F/hr	500
c.	Loading, 5%/min.	15000
d.	Unloading, 5%/min.	15000
e.	Step Load Increase, 10%	2000
f.	Step Load Decrease, 10%	2000
g.	Reactor Trip or Loss of Load	400
h.	Hydrostatic Test, 3125 psia, 100-400°F	10
i.	Leak Test, 2250 psia, 100-400°F	200
j.	Normal Plant Variations, ± 100 psi, ± 6°F	10^6
k.	Pump Starting and Stopping	4000
l.	Loss of Flow	40
m.	Abnormal Loss of Load (Turbine Trip with delayed reactor trip)	40
n.	Loss of Secondary Pressure	5

The preceding provides a summary of how cyclic loads due to plant operating conditions are accounted for. It is not all inclusive, nor is it intended to be. Rather, it is provided to indicate how these loads are accounted for and to confirm that the design can accommodate appropriate operating transients.

3.9.2.3 Code Case Interpretation

The code case interpretations of the ASME and ANSI codes that were incorporated in the addenda in effect at the time of purchase of the equipment are the interpretations which were referred to regarding these components.

Per Reference 28, for repair or replacement of Class 2 piping that cannot be isolated by existing valves or that require securing safety or relief valves for isolation, the pressure test required by IWA-4400 will be deferred until the next regularly scheduled system hydrostatic tests (IWC-5000), with the repaired or replaced piping subject to the following conditions:

a) prior to or immediately upon return to service, a visual (VT-2) examination for leakage shall be conducted during a system functional test or during a system in-service test in the repaired or replaced portion of the piping system; and

b) the repair or replacement welds shall be examined in accordance with IWA-4000 and IWA-7000 using volumetric examination methods (IWA-2230) for full penetration welds or surface examination methods (IWA-2220) for partial penetration welds.

Code Case N-411, "Alternative Damping Values for Response Spectra of Class 1, 2 and 3 Piping, Section III, Division 1," may be applied to new systems analyzed by response spectrum methods. Code Case N-411 may also be utilized to qualify proposed modifications to existing systems. Based on the use of Code Case N-411, all piping qualification analyses shall include verification that:

- All piping supports are properly designed and capable of withstanding design loads.
- Excessive pipe deflections are not introduced by use of the code case (i.e., displacements shall be checked to verify that proper clearances exist with respect to adjacent structures, components and equipment). Pipe mounted equipment shall also be checked to assure that the equipment is able to withstand the pipe motion.
- Postulated pipe break locations have been properly considered.
- Affected equipment nozzle loads are not adversely affected.

Each new analysis or reanalysis performed utilizing the PVRC damping values shall include specific reference to Code Case N-411 in the Quality Assurance Records associated with the calculation. For each anchor group (analysis package) where the code case is applied, the code case shall be applied to the entire analysis (i.e., PVRC damping would not be mixed in a given analysis with Regulatory Guide 1.61 criteria).

Code Case N-411 may be applied to systems analyzed by response spectrum methods.

3.9.2.4 Active Components – Code Classes 1, 2 and 3

For a faulted system, components that are associates with the faulted system only and do not perform any isolation function need only retain the pressure containing boundary, i.e., they should remain reasonably intact but need not satisfy operability related criteria. Unit 1 equipment specifications specifies represented upper limit type pressure and temperature conditions for the component. The active safety class pumps and valves are specified to withstand concurrent pressure temperature, and seismic loads with no loss of function.

Mechanical components in fluid systems whose operability is required to perform a safety function are considered active components. The operability of these components must be assured under the loading conditions they will be subjected to during a postulated accident including the design basis earthquake. Pumps and valves are the only active components relevant to St. Lucie 1. The operability assurance program for both pumps and valves is described below:

a) Loading conditions are evaluated and conservative loading requirements based on the most adverse combination of loads are included in component specifications.

b) Specifications require compliance with the Draft ASME Code for Pumps and Valves - November 1968.

c) Vendors must demonstrate by analysis and/or testing that components will not suffer loss of function when seismic loads are imposed in addition to other applicable loads.

d) Vendor tests are performed to demonstrate integrity and performance capability.

e) Construction and preoperational checks and tests verify proper installation and in situ performance.

f) Periodic tests during the operating life of the facility monitor insitu performance during service life.

g) The St. Lucie quality assurance program insures that components are designed, manufactured, installed, and tested in accordance with applicable codes, regulatory requirements, and component specifications.

h) A periodic in-service inspection program will monitor component integrity during service life.

During components design, evaluation limitations are placed on stress levels that may be attained and areas where deformation might affect performance are identified and evaluated. To ensure that active components remain functional during all loading conditions, including

DBE, stresses are limited to within 90 percent of yield stress. Stress calculations, rather than empirical methods, are used to verify that the stress criteria are met.

With regard to deformations, the general approach is to identify areas or parts of active components where significant (local) deformation might impede the free movement of rotating or reciprocating parts.

These areas or parts are analyzed to confirm that the stress levels achieved in their design are low enough that the associated deformation would be correspondingly small. This is responsive to the guidance provided by Regulatory Guide 1.48 for active components subjected to loadings which include seismic. For example:

a) For Classes 2 and 3 active pumps, the significant areas selected for evaluation were typically the foundation bolting and bearing bracket bolting.

b) For Classes 1, 2, or 3 active valves, the most significant area is the yoke. For motor operated active valves, a curved beam analysis was performed which established the stress in a very localized area. The maximum value reported was about 2/3 of the material specification minimum yield. The overall deformation associated with this level of stress in the very localized area is very small.

Nozzle loadings are taken into account within normal design practice. Generally, the following precautions are taken to ensure that the end loads applied to active pumps do not effect normal operation:

a) Component specifications for active pumps require that the pump manufacturer submit the values of the maximum allowable forces and moments for the pump nozzles.

b) These end loads are established by the manufacturer to very conservative levels such that the pump will operate normally. A statement to this effect has been obtained from the manufacturer.

The Architect-Engineer applies the pump manufacturer supplied end loads as limits for designing the piping system for the faulted plant condition. In one instance this exchange of information caused the LPSI pump manufacturer to make installation modification recommendations for two pumps where the Architect-Engineer was unable to reduce pump nozzle loading by changing piping layouts.

Valves are stronger than the pipe associated with them. They are designed for the pressure, temperature, and seismic load associated with the faulted condition. Thus, operability, is only a concern in the faulted condition where a moment is applied to the valve body. For those valves where (i) the valve must operate to achieve an isolation function, and (ii) pipe restraints are not supplied to eliminate the moment, the manufacturers of these active valves will be requested to stipulate that each valve will operate normally when subjected to the end connection loads associated with faulted plant conditions.

The St. Lucie 1 operability assurance program provides adequate assurance that active components will function when called upon to do so. In addition, the component redundancy provided in the design accommodates single active random component failure in a safety system without loss of required system performance levels. Active component operability assurance is a continuing industry program. Active components, which are common to all nuclear facilities, are continually being analyzed by Applicants.

An illustration is given below of an application of the operability assurance program. Some of the above features in the specifications for the main steam isolation valves and the containment spray pumps are:

a) Main Steam Isolation Valves

1) Seismic Design -

i) Valves and their operators and appurtenances shall be able to withstand seismic forces of 0.60g horizontal and 0.375g vertical (Operating Basis Earthquake). The vertical and horizontal forces shall be assumed to be acting simultaneously.

ii) The design basis seismic forces (DBE)to be withstood are 1.20g horizontal and 0.75g vertical. The vertical and lateral forces shall be assumed to be acting simultaneously.

iii) The unit stresses obtained from seismic loads shall be added directly to unit stresses from all other loadings. The allowable unit stresses shall not be increased due to the addition of the OBE seismic load. The allowable unit stresses may be increased due to the addition of the DBE seismic load to a limiting value that will cause no loss of function due to the seismic loads.

iv) Seller shall supply calculations which substantiate that the valves and their components will not suffer loss of function due to the addition of seismic loadings.

2) Manufacturers' Tests

i) Valves shall be hydrostatically tested at the pressure standard as per ANSI Standard B16.5.

ii) Valve seats shall be given the standard 80 lb air test in accordance with the Manufacturer's Standardization Society Standard Practice SP-61 for the hydrostatic testing of steel valves.

iii) Seat leakage rate shall not exceed 2 cc/hr per inch of diameter across the seat when fully closed and under full hydrostatic test pressure.

iv) Stop valves shall be tested for backseating tightness to demonstrate that they can be repacked under full working pressure.

v) Mechanical tests shall be performed on stop valves for opening and closing response times.

vi) If vibrating platform tests are performed, they shall be conducted at various frequencies over the expected range associated with the DBE.

vii) Operators shall be completely assembled onto their respective valves and the entire valve shall be shop tested for operability before shipment.

viii) All valve body castings shall be 100 percent radiographed after heat treatment.

ix) All external accessible internal surfaces of the valve fluid boundary containment shall be examined by magnetic particle methods including final machined surfaces.

x) Mill test reports containing physical and chemical analysis sheets on fluid containing parts of valves shall be provided.

3) Unique Service Condition -

The valves shall be suitable for outdoor service in a salt laden atmosphere; be subject to hurricane winds of 194 mph, or tornado winds of 360 mph; torrential rains; and high ambient temperature and humidity.

b) Containment Spray Pumps

1) Seismic Design -

i) Loading conditions considered in the design of the pumps shall include:

a- Internal pressure and mechanical forces including forces due to fluid flow.

b- Temperature transients

c- Earthquake shock as follows:

ii) The OBE seismic load shall be considered as a lateral force equal to 0.187g times the permanent gravity loads and a vertical force acting either up or down equal to 0.125g times the permanent gravity loads. The vertical and lateral forces shall be assumed to be acting simultaneously.

iii) The DBE seismic load shall be considered as a lateral force equal to 0.375g times the permanent gravity loads and a vertical force acting either up or down equal to 0.25g times the permanent gravity loads. The vertical and lateral forces shall be assumed to be acting simultaneously.

iv) The unit stresses obtained from seismic loads shall be added directly to unit stresses from all other loadings. The allowable unit stresses shall not be increased due to the addition of the OBE seismic load. The allowable unit stresses may be increased due to the addition of the DBE seismic load to a limiting value that will cause no loss of function due to the seismic loads.

v) Seller shall supply test data, operating experience and/or calculations which substantiate that the pumps will not suffer loss of function due to the addition of seismic loadings.

2) Manufacturers' Tests

   i) All pump test set ups, test procedures and instrumentation shall be in accordance with standards of the Hydraulic Institute and the ASME Power Test Code PTC-8.2.

   ii) Performance at all loads within the design capacity shall be smooth and free from undue noise, vibration, deflection, overheating and wear.

   iii) Hydrostatic test of the pump casing shall be performed on each pump at 150 percent of shut off total head prior to painting.

   iv) Complete performance test with at least five (5) head-capacity points shall be conducted in the shop to establish pump characteristics. Transient test shall also be conducted at the pump design point to establish pump ability to withstand different water temperatures. NPSH requirements for the capacity range should be verified by a suction pressure suppression test for each pump.

3) Unique Service Condition

   i) The pumps are initially expected to take suction from the refueling water tanks with minimum water temperature of 40°F and boron concentration of 0 to 1900 ppm. They would also take suction from the containment sump with an assumed maximum water temperature of 250°F. The pump design and construction shall be able to withstand this temperature transient occurring within 10 seconds of switchover signal.

Active components, which are defined as components carrying fluid required to undergo a mechanical motion to either mitigate the effects of an accident or safely shut down the plant are listed below. (Table 5.2-3 provides a tabulation of Class I active components.)

1.

PUMPS	TYPE	QUANTITY	QUALITY GROUP
Containment Spray	Vertical	2	B
Intake Cooling Water	"	3	C
Component Cooling Water	Horizontal	3	C
Auxiliary Feedwater (Motor)	"	2	C
" " (Turbine)		1	C
High Pressure Safety Injection	"	2*	B
Low Pressure Safety Injection	"	2	B
Charging	Positive Displacement	3	B
Diesel Oil Transfer	Horizontal	2	C

* 1C HPSI pump has been abandoned in place.

2. VALVES	TYPE	QUANTITY	QUALITY GROUP
Main Steam System			
S.G. Outlet	Air	2	B
	Check	2	B
M.S. to Aux.Turbine	Manual	4	B
	Check	2	B
	Motor	1	B
Feedwater System			
S.G. Outlet Feed	Check	2	B
	"	2	B
	Motor	2	B
Aux. Feed.to S.G.(motor)	Check	2	B
	Motor	4	C
	Check	2	C
Aux. Feed. to S.G.(tur.)	Check	2	B
	Motor	2	C
	Check	1	C
Cond. Tk. to Stm. Tur. Pump	Check	1	C
Intake Cooling Water Sys.			
Pump discharge to header	Check	3	C
Turbine cooling isolation valves	Motor	2	C
Header	Air	2	C
Strainer Debris Discharge	Air	2	C
Component Cooling Water Sys.			
Pump discharge	Check	3	C
Cross-connect	Air	2	C
To fan coolers	Motor	4	B
To reactor coolant pumps	Air	4	B
Outlet from SDHX	Air	2	C
Pump suction	Air	2	C
Primary Makeup Water Sys.			
Containment penetration	Motor	1	B
	Check	1	B
Instrument Air Sys.			
Containment penetration	Motor	1	B
	Check	1	B
To annulus accumulator	Check	5	B
S.G. Blowdown System			
Blowdown	Air	4	B
Blowdown sample	Air	4	B

2. VALVES	TYPE	QUANTITY	QUALITY GROUP
Containment Spray Sys.			
Pump Suction from RWT	Motor	2	B
	Check	2	B
Pump discharge	Check	2	B
	Air	2	B
	Check	2	B
RWT to LPSI pumps	Check	2	B
Pump Suction from cont sump	Motor	2	B
	Check	2	B
Safety Injection System			
SI Tank Discharge	Check	4	B
LPSI Header	Motor	4	B
HPSI Header	Motor	4	B
HPSI Header - Auxil.	Motor	4	B
LPSI Pump Suction	Check	2	B
LPSI Pump Discharge	Check	2	B
HPSI Pump Suction	Check	2	B
HPSI Pump Discharge	Check	2	B
HPSI Pump to Header	Motor	1	B
HPSI Pump to Header - Auxil.	Motor	1	B
Shutdown Cooling Heat Exchanger	Flow Control	1	B
Mini-recirc line	Motor	2	B
Reactor Coolant System			
RCS Gas Vent	Solenoid	4	B
Other Containment Isolation Valves	Refer to Table 6.2-16		

Consistent with this generation of units, calculations were performed to verify the design adequacy of components. Since the equipment is already shipped from the manufacturer's shops and has been installed, it is not possible to institute a testing program for Unit 1 equipment. However, it is feasible to test vital appurtenances on a type test basis. Tests have been conducted. The results of testing conducted to date are included in the discussion that follows:

The following appurtenances utilized on all Pump and Valve Code Active Class 1, 2 and 3 pneumatic operates control valves within CE, Inc. scope of supply have been tested by Fisher Controls Company:

a) Fisher type 67 FR Regulator

b) National Acme Snap Lock Electric Switch No. D240OX-2

c) Fisher Type 2625 Volume Booster

d) Fisher Type 546 Transducer

e) Fisher Type 3590 Positioner

f) ASCO solenoid 832OA22

The following testing was performed on all of the above referenced equipment and no malfunctions were detected:

a) Each appurtenance was tested in three orthogonal axes mounted to a vibration table using the standard mounting brackets of the appurtenance.

b) All tests were conducted with the appurtenance in an operating condition.

c) A continuous frequency sweep was run in each of the three axes at an acceleration level of 1.0g from 5 to 60 Hz. During the test the output of the appurtenance was continuously monitored. If no resonance was observed, a one-minute 4.0g dwell test was run at 10, 17, 25, and 33 Hz. In only one test was a resonance located below 60 Hz, that being a resonance at 53 Hz in one axis for the type 3590 positioner. A 4.0g dwell test was conduced on this positioner at the resonant frequency for one minute resulting in no malfunction.

d) Malfunction criteria were as follows:

   1) Loss of output

   2) Erratic or unwanted output ±5 percent of output span

   3) Major calibration shift ±5 percent of output span

   4) Structural failure, broken or loosened parts

All components performed satisfactorily. The ASCO solenoid (No. 8302C27R) utilized on all pneumatic operated valves in CE, Inc. scope has not been tested. Provision has been made with ASCO to have this valve tested to the requirements of IEEE-344.

As part of the requirement to demonstrate either by analysis or test that the overall valve assembly is capable of withstanding a 3.0g load applied in any direction, Fisher Control Co. fractured standard production cast iron yokes for all Pump and Valve Code Class 1 and 2 valves in CE, Inc. scope and demonstrated that in the weakest attitude, the lowest "G" load to fracture for any of these valve yokes was 10g for two valve designs out of the thirteen designs tested. Maximum g load capability on the eleven other designs ranges from 10.3 to 32.5g.

Valve motor operators supplied on St. Lucie Unit 1 were manufactured by Limitorque. These operators are representative of the prototype units that were successfully seismically tested in accordance with Ogden Technology Laboratories Report No. 7192-9 dated 9/26/72 and Lockheed Electronics Company Test Report No. 2120-4594 dated July 31, 1968, and No. 2539A-4723 dated September 23, 1973. These reports show that the operators comply with the intent of IEEE Standard 344-1971 requirements.

A Limitorque Model SMB-0-25 was vibration tested by Ogden Labs as follows:

An exploratory test was performed in three orthogonal axes over a frequency range of 1 to 35 Hz at a maximum acceleration of 1.0g to determine natural frequency. No natural frequencies were detected. The SMB-0-25 was vibrated at 3g's for ten seconds at all even numbered frequencies from 2 Hz to 34 Hz and at a level between 5.3 and 5.8g's at 35 Hz.

A Limitorque Model SMB-0-15 was vibration tested by Lockheed Electronic Co. as follows:

The SMB-0-15 was subjected to five simultaneous tests in both the vertical and horizontal axes. Each test consisted of two minutes of vibration at 35 Hz at an acceleration level of 3g's followed by one minute vibration. Vibration scans for natural frequency were also conducted in both axes of vibration between 5 and 35 Hz. No resonances were detected.

A Limitorque Model SMB-0-25 was vibration tested by Lockheed Electronic Co. as follows:

The SMB-0-25 was mounted on a test stand and having a threaded valve stem being driven by the Limitorque operator simulating opening and closing a valve. The Limitorque operator was electrically connected so as to stop at the full close position by means of a torque switch and stop at the full open position by means of a geared limit switch. The Limitorque operator had a 4-train geared limit switch installed and all contacts not being used for motor control were wired to electric indicating lights at a remote panel.

The unit successfully completed a 5.3g shock level at 35 Hz with no discrepancies noted. An exploratory scan of 5 Hz to 35 Hz was made and no critical resonant frequencies were noted on the Limitorque operator. The unit was shocked and vibrated in each of three different axes a total of 2 minutes on, 1 minute off, three times per axis. The unit was operated electrically to both the full open and full close position and all torque switches and limit switches functioned properly. None of the auxiliary limit switches wired to indicating lights ever flickered or indicated they were opening or flickering. All electrical and mechanical devices on the operator worked successfully.

An additional test level of 10g's at a maximum of 49 Hz was also conducted. The Limitorque operator had no defects during the first two minutes of operation at the 10g level. However, upon starting the second run, the hardware holding the geared limit switch loosened, and the test was discontinued. At that time, the unit had been subjected to a total of 9 minutes of shock and vibration at 10g's and 49 Hz.

NRC Generic Letter 89-10 requires that operating nuclear plants develop and implement a program to ensure that switch settings on all safety-related motor-operated valves (MOVs) are correctly selected, set and maintained to accommodate the maximum differential pressures expected on these valves during all postulated events within the design basis. Item a) of the Letter requires that the design basis for these MOVs be reviewed to determine the maximum differential pressure expected during both opening and closing strokes for all postulated events.

Item b) of Generic Letter 89-10 requires that the licensee establish the correct MOV switch settings based on the previously determined maximum differential pressure. All switches, including torque switches, torque bypass switches, position limit, position indication, overloads, etc., shall be considered. This requires that the actuator and valve capabilities at degraded voltage be evaluated. Modifications to the valves and actuators have been performed where appropriate to allow incorporation of the proper switch settings.

Once the correct switch settings have been incorporated into the respective motor-operated valves, Item c) of Generic Letter 89-10 requires that each motor-operated valve be stroke tested against the maximum differential pressure established in Item a) to verify operability. The results of these tests shall be used to trend the condition of the valves and operators and to help determine appropriate maintenance frequencies.

The requirements of NRC Generic Letter 89-10 have been completed for the applicable Unit 1 valves, which are listed in Table 3.9-6.

As a result of NRC Generic Letter 96-05, "Periodic Verification of Design- Basis Capability of Safety-Related Motor-Operated Valves," FPL committed to maintaining a periodic verification program to ensure safety related MOVs remain capable of performing their safety functions within the current licensing basis. The program addresses all the elements of Generic Letter 96-05 and the scope includes all valves within the Generic Letter 89-10 program. The basis of the program activities and frequencies, and justifications for any changes are not commitments and will be maintained at the St. Lucie plant. See Reference 30.

The following appurtenances are vital to the operation of the safety injection pumps, other CE supplied pumps have none.

a) Mechanical seals and bearing coolers

b) Piping associated with these coolers

The coolers are bolted rigidly to the plate of the pump assembly, and the piping is supported such that no significant vibration can take place. The weight of the cooler relative to the size of its support brackets and bolting precludes any possibility of high stress or deformation that might prevent these appurtenances from operating. Therefore, no provision for the testing of these appurtenances is considered necessary.

Pump motors have been reviewed. The vendor has certified that the category IE station auxiliary motor as well as appurtenances will withstand the seismic forces specified in the Unit 1 specifications. See Appendix 3B, item I.

3.9.2.5 Pipe Whip Protection

3.9.2.5.1 General Criteria

The basic design approach to protect critical systems from the effects of pipe whip is to provide separation between piping and equipment. Where separation is not possible due to physical limitations, pipe whip restraints are designed and installed. Design basis criteria are provided in Section 3.6.2.

3.9.2.5.2 Basic Assumptions

The design of pipe whip restraints is based on the following assumptions:

a) Pipe rupture, both circumferential (guillotine) and longitudinal (slot), can occur anywhere in the system.

b) Pipes which can only be pressurized from reciprocating pumps are not considered for creating jet impingement or reaction forces, since after a pipe rupture in such lines the pressure will decay immediately.

c) Pipes with maximum operating pressures of 125 psig or less are not considered. Typical calculations for such lines have shown that impingement and reaction forces are too small to create failures in adjacent components.

d) Protection of adjacent piping or equipment from pipe whip is provided by limiting the deflection of the pipe at the postulated rupture to small magnitudes. This has been achieved by placing pipe restraints such that the reaction forces on the broken pipe do not cause formation of a plastic hinge.

The systems postulated to rupture, outside the reactor coolant pressure boundary are provided in Section 3.6.1.

3.9.2.5.3 Method of Analysis

Pipe whip and jet impingement forces are evaluated and pipe restraint spacing is provided in accordance with the methods described in Section 3.6.4.

3.9.2.6 Mounting of Safety and Relief Valves

Installation of pressure relieving devices for the reactor coolant system and main steam system will be in accordance with ASME Boiler and Pressure Vessel Code, Section III. A description of the pressure relieving devices for the main steam system is given in Section 5.5.1.

The full design thrust of the pressure relieving devices on the main steam lines and main steam line header outside the containment is furnished by the manufacturer of the pressure relieving device. The bending, torsion, shear and direct stress is established for the pressure relieving device nozzle, valve nozzle, and the nozzle to header or pipe intersection in the cases where the device relieves directly to the atmosphere.

Calculated stresses in the nozzle neck include the effects of discharge loads, seismic effects, weight and pressure. Nozzle-header combinations are checked for full conformance with applicable paragraphs of ASME Boiler and Pressure Vessel Code Section III for all required load combinations including, thrust seismic loads, weight and pressure.

An additional check is made on the steam lines and header which considers the effects (including torsion) of all valves discharging at maximum thrust simultaneously.

For closed systems where the fluid is discharging from a safety relieving device to another vessel or chamber, both the discharge and inlet ends of the connecting pipe (which includes the pressure relieving device) are considered as anchors. The condenser and the pressurizer relief tank (discharge ends) and steam line, header and pressurizer (inlet ends) will be analyzed for this condition. The dynamic forces and static pressure forces throughout the connecting pipe are considered, and the resultant forces and moments in combination with seismic and other effects are used to determine the system stresses and required connecting pipe restraints.

3.9.2.7 Analytical Methods and Criteria

Safety related pumps were designed to commercial standards as required by the draft pump and valve code.

In the case of active valves, the criteria of the design codes and standards are supplemented to require design calculations which further demonstrate the structural adequacy of the component to remain functional during all specified loading conditions. In these cases, the primary stress produced in the components and supports by the combined effects of the most severe loading combinations, which includes coincident operating conditions plus DBE loadings, is limited to the material yield strength at temperature.

The draft pump and valve codes specify that design shall be in accordance with manufacturers standard practices. For each support system the specification gives the conditions of design (i.e., seismic, tornado, or hurricane plus operational loads). The manufacturer's calculations are reviewed to ensure that the specified conditions were considered in the equipment support design. The purchase specification requires that the allowable stresses specified in the ASME or AISC codes not be exceeded for the design condition (OBE plus operational loads). The stresses allowed for the more severe loading combinations are determined by the manufacturer according to his standard practice. The only criteria for these stress levels is that there be no loss of support function. An example is the component cooling water pumps where the conditions of design were seismic, tornado, or hurricane plus operational loads. The allowable stress for the design condition was from the AISC codes. The stresses allowed by the manufacturer for more severe conditions (DBE + operational, tornado + operational, and PMH + operational) were 1.33 times the allowable stress.

The design calculations performed to demonstrate the adequacy of the components and supports employed elastic methods. Specifically, plastic deformations are not permitted in active components or in the supports for active components.

Relief valve installations are analyzed for steady state load conditions. To evaluate reaction force, both a static pressure term and a momentum term due to discharge jet velocity are considered. For multi-valve installations, analysis is restricted to individual and simultaneous valve discharge.

All active safety related pumps are subjected to operational tests to demonstrate their ability to function in the design condition. The design conditions for these components are the conditions corresponding to accident conditions for the reactor coolant system. In addition, the high pressure safety injection pump design was tested under the severe thermal transient conditions which would occur in the unlikely event of a loss-of-coolant accident.

Components which are constructed in accordance with the equipment specifications are required to comply with the applicable codes as mentioned in the specifications. The manufacturer is responsible for compliance with the design specifications and must submit documentation that the materials and welds satisfy the design specifications by providing the results on one or more of the following tests: radiographic, liquid penetrant, magnetic particle, ultrasonic or hydrostatic. The seismic analysis or testing described in Section 3.9.1.2 provided by the manufacturer also serves to demonstrate compliance with the applicable sections of the codes.

3.9.2.8 Operational Cycles

The auxiliary feedwater pumps may be subjected to the following number of operational cycles during the plant life: testing 720 cycles in which the pumps run for 15 minutes during each test; plant cooldown, 500 cycles; and hot standby, 15,000 cycles. In all cases the electrically driven pumps are preferred for operation with the steam turbine driven pump on standby. However, the steam turbine driven pump may be subjected to 300 cycles of the complete system blacking out including the loss of the standby diesel generators. During the performance of the operation the motor operated valves on the discharge are kept closed and the pumps operated on the minimum recirc flow.

Both the electrically driven and the steam turbine driven pumps are capable of withstanding without any damage instantaneous loss of suction should this occur inadvertently.

The component coolant pumps are run continuously while the plant is in operation and may be subjected to 500 shutdown cooling cycles. One of the three component cooling pumps will be on standby at all times. Standby condition will be alternately shared among the three pumps.

The containment spray pumps are tested every refueling outage and thus will undergo approximately 40 lifetime full-flow testing cycles.

3.9.2.9 Storage Tanks

The condensate storage tank is designed to "American Water Works Association Standard for Steel Tanks, Standpipes, Reservoirs and Elevated Tanks for Water Storage", AWWA D-100.

The refueling water storage tank is designed to ANSI B96.1, standard for "Welded Aluminum Alloy Field Erected Storage Tanks".

The diesel oil storage tanks are designed to API 620, American Petroleum Institute Standard for Low Pressure Storage Tanks.

The load combinations and stress-limits for the condensate storage, refueling water storage, and diesel oil storage tanks are as follows:

	Load Combination	Stress Limit
1.	Dead + Live + Wind (1)	Basic code allow
2.	Dead + Live + Tornado (2)[1]	90 percent of the yield strength of the material
3.	Dead + Live + Operating Basis Earthquake (3)	1.33 of Basic code allow
4.	Dead + Live + Design Basis Earthquake (3)	90 percent of the yield strength of the material

(1) Normal wind load shall be in accordance with the South Florida Building Code except that the design pressure at any level shall not be less than 50 psi on the full projected area for the tank empty or full of water. No allowance shall be made for the shielding effects of buildings.

(2) Tanks are designed to withstand tornado wind of 360 mph. The force on the roof shall be assumed to be uniform internal pressure of 3 psi. Tanks are assumed at least 90 percent full of product.

(3) Tanks are designed for concurrent horizontal and vertical seismic loads. Results of dynamic analysis take into account sloshing effects obtained from hydrodynamic analysis. For roof design, the vertical seismic force at the roof elevation.

Notes

1. Load combinations including tornado loads are not applicable to the diesel oil storage tanks. UFSAR Appendix 3F Section 4.3.4, 5.0 and 6.7 discuss the inherent adequate tornadic debris capability of the diesel oil storage tanks and the further enhancement of the tornado resistance due to the missile protected intertie to the Unit 2 diesel oil storage tanks. The Unit 1 diesel oil transfer pumps and the Unit 2 diesel oil storage tanks are missile protected.

3.9.2.10 Supports for ASME Class 2 and 3 Components

Nominal allowable stress limits for supports, from draft ASME Pump and Valve Code, or AISC as applicable, were supplied to all loading conditions for Class 2 and 3 pumps without increase permitted for seismic or accident loadings. Class 2 and 3 pumps are bolted directly to the floor without any intermediate supports.

Support systems for code class 2 and 3 piping and components were designed to restrict stresses to within the applicable code allowable for normal, upset, emergency and faulted plant operating conditions. The "limiting" stress conditions for pipe runs and connected components formulated the basis for design.

A detailed stress analysis for the "limiting" plant operating conditions was performed for each pipe run to assure that the piping was not overstressed, to determine the maximum expected thermal displacement of the lines, determine the maximum expected loadings at all pipe terminal points and define the actual loads at hanger and restraint points.

The maximum expected loads at the terminal ends (e.g., code class 2 and 3 pumps, tanks, etc.) are transmitted to the vendor who determines the equipment anchor bolt requirements. Operational, seismic and component connected piping loads were considered. Wind and hydrostatic loadings in addition to the above were considered for outdoor components when applicable.

Where possible, hanger and restraint functions were combined to eliminate redundant supports.

In addition, the following general design criteria were employed:

a) spring hangers were designed for the operating weight of the pipe

b) rigid restraints, both horizontal and vertical, were designed for thrusts due to operating load, thermal and seismic forces

c) snubbers were designed for thrusts due to seismic forces and/or fluid transient forces

d) pipe rupture restraints were designed for jet impingement and reaction forces. A minimum restraint gap of 1-inch was maintained to ensure that pipe rupture restraints do not interact during any condition other than pipe rupture.

3.9.2.11 Inservice Testing of Pumps and Valves

A pump and valve inservice test program for St. Lucie Unit 1 is in effect. The first ten year interval program which ended February 11, 1988, was conducted in accordance with the 1974 Edition of the ASME Code (through Summer 1975 Addenda) except for specific relief requested in accordance with 10 CFR 50.55a (g)(5)(iii). See References 22 and 27. The second ten year interval inservice test program began on February 11, 1988. The second ten year inservice pump test program was conducted in accordance with Subsection IWP, Section XI, Division 1, of the 1983 Edition of the ASME Boiler and Pressure Vessel Code through Summer 1983 Addenda, except for specific relief requested in accordance with 10 CFR 50.55a (f)(5)(iii). The third ten year interval began on February 11, 1998, and is being conducted in accordance with the requirements of the ASME B & PV Code Section XI, 1989 Edition and ASME/ANSI OM-Code, including OMA-88.

3.9.3 COMPONENTS NOT COVERED BY ASME CODE

The summaries of the stress and dynamic analyses of the reactor internals, given in Section 3.7.3.3 and in Reference 18 under LOCA conditions, demonstrate that all design loading combinations will be sustained without impairment of structural and functional capability. Details of the mechanical design and analytical procedures for the design of the fuel assemblies are discussed in Section 4.2.1.

REFERENCES FOR SECTION 3.9

1. Ghosh S., Wilson E., "Dynamic Stress Analysis of Axisymmetric Structures under Arbitrary Loading," Dept. No. EERC 69-10, University of California, Berkeley, September 1969.

2. Kiss, E., "Analysis of the Fundamental Vibration Frequency of a Radial Vane Internal Steam Separator Structure," from the Proceedings of the Conference on Flow Induced Vibrations in Reactor System Components, May 1970, ANL-3685, p. 335-343.

3. Hermann G., Mirshy I., "Three Dimensional Shell Theory Analysis of Axially Symmetric Motions of Cylinders," Journal of Applied Mechanics, Trans. ASME, Vol. 78, p. 563-568, 1956.

4. Flugge W., "Stresses in Shells," Third Printing, Springer-Verlag, New York, 1966.

5. Koval L. R., Cranch E. I., "On the Free Vibrations of Thin Cylindrical Shells Subjected to Initial Torque," Proceedings of the U. S. National Congress of Applied Mechanics, p. 11, 1962.

6. Reismann H. and Padloy J., "Forced, Axi-Symmetric Motions of Cylindrical Shells," Journal of the Franklin Institute, Vol. 284, No. 5, November, 1967.

7. Tang Sing-Chih, "Response of A Finite Tube to Moving Pressure," Journal Engineering Mechanics Division, ASCE, Vol. 93, No. EM3, June, 1967.

8. Tillerson J. R., Haisler W. E., "SAMMSOR II - A Finite Element Program to Determine Stiffness and Mass Matrices of Shells-of-Revolution," Texas A&M University, TEES-RAPT-70-18, October, 1970. "DYNASOR II - A Finite Element Program for the Dynamic Non-Linear Analysis of Shells-of-Revolution," Texas A&M University, TEES-RPT-70-19, October, 1970.

9. Klein S. and Sylvester R. J., "The Linear Elastic Dynamic Analysis of Shells of Revolution by the Matrix Displacement Method," Air Force Slight Dynamics Laboratory, TR-66-80, 1966, p. 299-329.

10. Johnson D. E., Greif R., "Dynamic Response of a Cylindrical Shell: Two Numerical Methods," AIAA Journal, Vol. 4, No. 3, March, 1966, p. 486-494.

11. Huang N.C., "Axisymmetric Dynamic Snap-through of Elastic Clamped Shallow Spherical Shells," AIAA Journal, Vol. 7, No. 2, February, 1969, p. 215-220.

12. Stephen W. B. and Fulten R. E., "Axisymmetric Static and Dynamic Buckling of Spherical Caps due to Centrally Distributed Pressures," Paper 69-89, AIAA Journal, 1969.

REFERENCES FOR SECTION 3.9 con't

13. Lock M. H., Okrebo S., and Whittier J. S., "Experiment of the Snapping of a Shallow Dome Under a Step Pressure Loading," AIAA Journal, Vol 6, No. 7, July 1968, p.1320-1326.

14. Stricklin J. A. and Martinez J. E., "Dynamic Buckling of Clamped Spherical Caps Under Step Pressure Loadings," AIAA Journal, Vol 7, No. 6, June 1969, P.1212-1213.

15. Kotanchik J. J., et al, "The Transient Linear Elastic Response Analysis of Complex Thin Shells of Revolution Subjected to Arbitrary External Loadings, by the Finite-Element Program SABOR 5 - DRASTIC," AD-709-189, Massachusetts Institute of Technology, April 1970.

16. Klein S., "A Static and Dynamic Finite Element Shell Analysis with Experimental Verification," International Journal for Numerical Methods in Engineering, Vol 3, p. 299-315, 1971.

17. Cabrielson V. K., "SHOCK - A Computer Code for Solving Lumped-Mass Dynamic Systems," SCL-DR-65-34, January 1966.

18. CENPD-42 (Proprietary) Topical Report on Dynamic Analysis of Reactor Vessel Internals Under Loss-of-Coolant Accident Conditions With Application of Analysis to C-E 800 MWe Class Reactors.

19. CENPD-55 "Analysis of Flow-Induced Vibrations Maine Yankee Precritical Vibration Monitoring Program Predictions," May 1972.

20. CENPD-85 "Analysis of Flow-Induced Vibrations, Fort Calhoun Precritical Vibration Monitoring Program Predictions," January, 1973.

21. CENPD-93 "Maine Yankee Precritical Vibration Monitoring Program Final Report," February 1973.

22. R. E. Uhrig (FPL) to R. W. Reid (NRC) Re: St. Lucie Unit 1 Docket No. 50-335, Inservice Test Program, L-80-136 dated May 1, 1980.

23. J. W. Williams (FPL) to J. R. Miller (NRC) Re: St. Lucie Unit 1 Docket No. 50-335, Reactor Vessel Internals and Thermal Shield: Plan Recovery Program-Final Integrity and Stability of Internals-Conclusions and Findings, L-84-29 dated February 10, 1984.

24. J. R. Miller (NRC) to J. W. Williams (FPL) Re: Thermal Shield Recovery Program dated March 14, 1984.

25. R. E. Uhrig (FPL) to R. A. Clark (NRC) Re: St Lucie Unit 1 Plant Recovery Program L-83-369 dated June 23, 1983.

26. J. W. Williams (FPL) to J. R. Miller (NRC) Re: St. Lucie Unit 1 Draft Failure Mechanism Analysis (FMA) L-83-531 dated October 25, 1983.

27. J. W. Williams Jr. (FPL) to J. R. Miller (NRC) Re: St Lucie Unit 1 Docket No. 50-335, Inservice Test Program, L-84-179 dated July 18, 1984.

REFERENCES FOR SECTION 3.9 con't

28. C. 0. Woody (FPL) to A. C. Thadani (NRC) Re: Alternative Rules for Hydrostatic Testing of Repair or Replacement of Class 2 Piping Section XI, Division 1, L-86-218 dated May 28, 1986.

29. J. A. Norris (NRC) to C. 0. Woody (FPL), dated August 17, 1989. "Closure of Generic Letter 89-08, Erosion/Corrosion Induced Pipe Wall Thinning."

30. J. A. Stall (FPL) to Document Control Desk (NRC), Re: 180-Day Summary Description, Generic Letter 96-05, L-97-61 dated March 11, 1997.

31. Westinghouse Calculation Note CN-AO-08-42 “St. Lucie Unit 1 EPU Transient Evaluation for the RCS Components Stress Intensities Design Bases,” Revision 1.

TABLE 3.9-1

MANUFACTURER'S SEISMIC ANALYSIS FOR CLASS II AND III EQUIPMENT

SPECIFIED SEISMIC CONSTANTS

Equipment	Operational	Design	Manufacturer Analysis
Auxiliary feedwater pump	*(h) 0.25g **(v) 0.175g	0.50g 0.334g	Stress calculation Based on the specified seismic constants
Component cooling pump and heat exchanger	(h) 0.40g (v) 0.25g	0.80g 0.54g	Stress calculations Based on the specified seismic constants
Containment spray pumps	(h) 0.187g (v) 0.125g	0.375g 0.250g	Stress Calculation Based on the specified seismic constants
Containment spray nozzles	(h) - (v) -	3.00g 2.00g	Nozzle was subjected to accelerations of 3.0g (h) and 2.0g (v) by a vibration test machine. Also, the nozzle was vibrated for one minute at each integral frequency from 5 Hz to 50 Hz
Intake cooling pumps	(h) 0.525g (v) 0.35g	1.05g 0.70g	Stress calculation Based on the specified seismic constants
2½ inch and smaller steel station valves	(h) 1.50g (V) 1.00g	3.00g 2.00g	Stress calculations
2½ inch and larger steel station valves	(h) 1.50g (v) 1.00g	3.00g 2.00g	Stress calculations Based on the specified seismic constants
Butterfly valves	(h) 1.50g (v) 1.00g	3.00g 2.00g	Stress calculations Based on 5g's simultaneously applied on valve components in each of three mutually perpendicular directions.

*(h) "g" acceleration acting in horizontal direction.
**(v) "g" acceleration acting in vertical direction.

TABLE 3.9-1 (Cont'd)

Equipment	Operational (1)	Design (1)	Manufacturer Analysis
Safety and relief valves	(h) 1.50g (v) 1.00g	3.00g 2.00g	Vibration test approved
Pumps			
charging	_	.6/.4	Vendor calculations based on specified seismic constants
other CVCS	_	1.5/1.0	
engineering safeguards	_	.26/.18	
waste management (2)	_	1.5/1.0	
waste gas comp. (2)	_	.36/.24	
Tanks			
all code tanks	.5/.33	1.01/.66	Vendor calculations based on specified seismic constants
ion exchangers	_	.5/.33	CE calculations based on specified seismic constants with vendor concurrence
filters	.24/.16	.48/.32	Vendor calculations based on specified seismic constants
Heat exchangers			
letdown	.11/.073	.22/.146	Vendor calculations based on specified seismic constants
regenerative	.11/.073	.22/.146	
shutdown	.11/.073	.22/.146	
Valves			
safeties & reliefs	_	3.0/3.0	Vendor calculations based on specified seismic constants
check & manual	_	3.0/3.0	
motor operated	_	3.0/3.0	
pneumatic control	_	3.0/3.0	
diaphragm	_	3.0/3.0	

(1) In most cases, only the design base seismic constants are specified since the higher acceleration and the yield stress limit are a more severe case than the operating base accelerations and the allowable stress limit.

(2) These components are not seismically qualified (Cat.I) per PC/M 82061.

TABLE 3.9-2

COMPARISON OF STRUCTURAL AND HYDRAULIC DESIGN PARAMETERS

For Historical Information

			Fort Calhoun	Maine Yankee	St. Lucie
	R_{mean},	in.	61-5/16	75-1/4	75-1/4
Upper CSB	t,	in.	2	2-1/2	2-1/2
	L,	in.	101-3/8	135-5/8	135-5/8
	R_{mean},	in.	61-1/16	74-7/8	74-7/8
Middle CSB	t,	in.	1-1/2	1-3/4	1-3/4
	L,	in.	166-1/8	144-3/4	144-3/4
	R_{mean},	in.	60-11/16	74-5/8	74-5/8
Lower CSB	t,	in.	2-1/4	2-1/4	2-1/4
	L,	in.	35-5/8	38	38
Lower	Cyl. ID,	in.	Integral	141	141
Core	Cyl. OD,	in.	Integral	145	145
Support	Cyl. L,	in.	Integral	42	42
Structure	Supported		Integral	CSB Flange	CSB Flange
Core Shroud Support			Bolted to CSB	Bolted to Core Support Plate	Bolted to Core Support Plate
	R_{mean},	in.	59-1/16	72-5/8	72-5/8
Cyl.	t,	in.	1-1/2	2	2
UGS	L,	in.	24	24	24
Beams,		in.	24 x 1-1/2	24 x 1-1/2	24 x 1-1/2
Plate t,		in.	3-1/4	4	4
Thermal Shield			Yes	Yes	Yes
No. of Loops			2	3	2
Design Min. Flow, 10^6 lbm/hr			71.7	122	122
Inlet Design Temp., °F			547	546	544
Inlet ID, in.			28-3/4	39	35-3/16
Outlet ID, in.			37	39-5/8	48-1/8
Inlet Pipe Vel, ft/sec			33.7	39.0	36.5
Downcomer Vel, ft/sec			25.2	24.9	23.4
Core Inlet Vel, ft/sec			12.8	12.9	13.5
Outlet Pipe Vel, ft/sec			41.5	39.0	40.8

CSB = Core Support Barrel
UGS = Upper Guide Structure
Vel.= Design Minimum Vel.

TABLE 3.9-2a

NORMAL OPERATING STATIC HYDRAULIC LOADS
ON THE CORE SUPPORT BARREL

Type of Load	Loading Value	Loading Condition
Axial Uplift Load	267,000 lb.	(See Table 8.3-4) Condition No. 2
Radial Pressure Differential Across CSB Wall at Thermal Shield Lug Elevation	29.9 PSI (Directed Radially Inwards)	Condition No. 1

TABLE 3.9-2b

NORMAL OPERATING CONDITIONS
FOR CALCULATING HYDRAULIC LOADS

Parameter	Condition No. 1 For Maximum Loading	Condition No. 2 For Minimum Loading
Inlet Temp	500°F	546°F
System Flow Rate	130% of PDES	115% of PDES
Power Level	Zero Power	2700MWt

Design flow range is based on a ~ 7% band about the best estimate measured (Cycle 1) flow rate of 123% of PDES (324,300 gpm).

TABLE 3.9-2c

PRESSURE FLUCTUATIONS

AT THE INLET NOZZLE STATION

T = 548°F

Pump Characteristic Frequency		Peak Pressure Fluctuation, P_{inlet}
Description	Value	Psi
1. Rotor Speed	15	± 0.15
2. 2 x Rotor Speed	30	± 0.05
3. Blade Passing	75	± 0.64
4. 2 x Blade Passing	150	± 0.08

TABLE 3.9-3

PIPING LOAD COMBINATIONS

Loading Condition	Code Class 1 Stresses	Code Class 2 and 3 Stresses
1. Design	$P_m \leq S_m$ $P_L + P_b \leq 1.5\ S_m$	Hoop stress≤S $S_e \leq S_A$ where S_A = f(1.25 S_c + .25 S_h)
2. Normal[(2)] (press + Wt)	$P_L + P_b + P_e + Q \leq 3.0\ S_m$ [(1)] $P_L + P_b + P_e + Q + F$ = S_p (use fatigue curve) [(3)]	Long stress ≤ S_h
3. Upset [(2,4)] (press + Wt + OBE + VT)	$P_L + P_b + P_e + Q \leq 3.0\ S_m$ [(1)] $P_L + P_b + P_e + Q + F$ = S_p (use fatigue curve) [(3)]	Long stress ≤ 1.2 S_h
4. Emergency[(2)] (press + Wt + DBE)	Max pressure ≤ 1.5 design pressure $P_L + P_b \leq 2.25\ S_m$	Max pressure≤1.5 design pressure Long stress ≤ 1.8 Sh
5. Faulted (2,5) (press + Wt + DBE + Rupture)	Max pressure ≤ 2.0 design pressure $P_L + P_b \leq 3.0\ S_m$	Max pressure ≤ 2.0 design pressure Long stress ≤ 3 S_h

(1) 3.0 S_m may be exceeded in accordance with the rules of Para. 1-705.4 ANSI B 31.7.

(2) As defined by ANSI B 31.7 Case 70, January 1970

(3) $S_{alt} = K_E \frac{SP}{2}$, cumulative usage factor ≤ 1

(4) Valve transients (VT) as appropriate. These include relief valve opening, relief valve closure, and fast valve closure (fast valve closure is not concurrent with OBE).

(5) Dynamic loads associated with pipe rupture (jet impingement).

TABLE 3.9-3A

LOADING COMBINATIONS ASME CODE CLASS 2 AND 3 COMPONENTS

Loading Combination	Class 2 or 3 Vessels Pressure Boundary	Class 2 or 3 Vessels Supports [5]
1. Design + OBE	$Pm \leq Sm^{(6)}$ Pm (or P1) + $Pb \leq 1.5\ Sm^{(6)}$	AISC Allowables
2. Normal + DBE	Same as (1)	Max. Tens < Yield Max. Shear < 1/2 Yield

	Class 2 Valves Pressure Boundary	Class 2 Valves Yoke	Class 3 Valves Pressure Boundary
1. Design	(1)	-	(1)
2. 3.0g Earthquake	-	Max. Tens < Yield	-

	Class 2 Pumps Pressure Boundary	Class 2 Pumps Support Bolts	Class 3 Pumps Pressure Boundary	Class 3 Pumps Support [5] Bolts
1. Design	(2)	-	(2)	-
2. Earthquake (values vary with the pump)	-	AISC Allowables	-	AISC Allowables

(1) Valves meet applicable requirements of draft Pump and Valve Code. Analyses were performed as required by that code. The primary pressure rating, P_r, is not exceeded for any condition, including the plant faulted condition.

(2) Manufacturers design criteria demonstrated by experience to be satisfactory for the specified design conditions. These components were ordered before issuance of the Pump and Valve Code. They were designed in accordance with the manufacturer's standard methods taking into consideration the parameters, as applicable, of operating pressure, temperature, seismic transients, and environmental loads that would be experienced by the equipment during plant faulted conditions. Acceptable pipe loadings were established in the pump specification and used by the pump vendor and the piping designer to limit nozzle loads. Testing of the pumps included significant loadings where applicable, e. g. transient thermal testing of first of a kind safety injection pumps. The seismic loadings of the safety related pumps was evaluated and determined to be non-significant considering the pump locations, seismic response spectra, pump configuration and method of support.

(3) Design measures such as pipe restraints, barriers and component isolation are provided, where necessary, to limit the loadings during the plant faulted condition to values not exceeding those applied in design. It must be noted that components within faulted systems that are not required to function subsequent to loss of the system need only retain their pressure retaining boundary.

(4) These supplementary design loading combinations and design stress criteria are applied and evaluated independently of all other design conditions.

TABLE 3.9-3A (con't.)

(5) Loading conditions, i.e., Seismic, Tornado or Hurricane (as appropriate) plus normal operating loadings are considered. Allowables employed by the component manufacturer for support materials varied from 1.33 normal allowable stresses to yield stresses as listed in ASME or AISC codes.

(6) Loading combinations shown are for the original equipment design Codes. The new Unit 1 ICW/CCW Strainers installed by PC/M 02025 utilize later NRC approved editions of the ASME Code along with St. Lucie Unit 2 defined load combinations.

TABLE 3.9-3B

CORE SUPPORT BARREL MIDDLE CYLINDER CODE ANALYSIS RESULTS

Normal Operation Plus Upset Conditions

Stress Category	Calculated Stress* psi	Allowable Stress psi
P_m	6,100	16,100
$P_m + P_b$	8,100	20,700
$P_m + P_b + Q$	23,800	48,300
Fatigue Usage Factor < 1		

* Includes Normal Operating Pressure plus OBE

Faulted Condition

Stress Category	Calculated Stress** psi	Allowable Stress psi
P_m	8,500	38,600
$P_m + P_b$	43,800	49,700

** Includes SSE plus LOCA

P_m = Membrane Stress

P_b = Bending Stress

Q = Secondary Stress

TABLE 3.9-4

DESIGN LOADING CONDITIONS

1. AUXILIARY FEEDWATER SYSTEM

A. PUMPS

1. Design

	Motor Driven	Turbine Driven
Quantity	2	1
Capacity, gpm	325	600
Head, ft	2660	2600
Fluid	water	water
Temperature		
Maximum, F	120	120
Minimum, F	40	40

2. Loading Conditions

Internal pressure and mechanical forces

Nozzle loadings

Tornado winds of 360 mph and 3.00 psi depression in 3 seconds

Hurricane winds of 194 mph and 1.5 psi depression

3. Code

ASME Section III, Code Class 3

B. PIPING

1. Design

Pump Suction

Pressure, psig	100
Temperature, F	200

Pump Discharge

	Motor Driven	Turbine Driven
Pressure, psig	1465	1420
Temperature		
Maximum, F	500	500
Minimum, F	150	150

Material - Carbon Steel ASTM A106 Gr B

4 in. and 6 in. - Schedule 80

2 in. and 1-1/2 in. - Schedule 160

2. Loading Conditions

Internal Pressure and Dead Weight Seismic Loadings

3. Code

B 31.7 Code Class 2 & 3

TABLE 3.9-4 (Cont'd)

1. AUXILIARY FEEDWATER SYSTEM (Cont'd)

C. VALVES[1]

1. Design
   Pump Suction - 150 lb ANSI rating
   Pump Discharge - 600 lb ANSI rating
   Material - Body and Bonnet - Carbon Steel

2. Loading Conditions

   Same as piping

1a. DIESEL GENERATOR FUEL OIL SYSTEM

A. PUMPS: Quantity 2

1. Design

   Pump Discharge - 30 psig

   Capacity - 25 gpm

   Fluid - Diesel oil

2. Loading Conditions

   Internal pressure and dead weight

   Nozzle loadings from connected piping

   Tornado winds of 360 mph and 3.0 psi depression in 3 sec.

   Hurricane winds of 194 mph and 1.5 psi depression

   Seismic loadings

3. Code

   ASME Section III, code class 3

TABLE 3.9-4 (Cont'd)

1a. DIESEL GENERATOR FUEL OIL SYSTEM (Cont'd)

B. STORAGE TANKS

1. Design

   Capacity - 20,593 gallons

   Material - Carbon Steel

   Pressure - Atmospheric

   Temperature - 125°F

2. Loading Conditions

   Internal pressure and dead weight

   Nozzle loadings from connected piping

   Hurricane winds of 194 mph and 1.5 psi depression

   Seismic loadings

3. Code

   API 620

C. PIPING & VALVES[1]

1. Design

   Suction - 150 psig

   Discharge - 150 psig

   Material - Carbon Steel

2. Loading Conditions

   Internal Pressure and Dead Weight Seismic Loadings

3. Code

   ASME Section III, code class 3

TABLE 3.9-4 (Cont'd)

2. COMPONENT COOLING SYSTEM

A. PUMPS: 3 Required

1. Design

TDH, ft - 177

Temperature
Maximum, F - 185
Minimum, F - 65

Capacity, gpm - 8500

Fluid - Demineralized water

2. Loading Conditions

Internal pressure and mechanical forces

Nozzle loadings winds of 360 mph and 3.00 psi depression in 3 seconds

Hurricane winds of 194 mph and 1.5 psi depression

Seismic Loadings

3. Codes

Standards of the Hydraulic Institute

ASME Section VIII and IX

NPVC Class III

B. HEAT EXCHANGERS

1. Design

	Tube	Shell
Pressure, psig	90	150
Temperature, F	150	185
Material	Aluminum - Brass SB-111 Alloy No. 687	ASTM A515, Gr 70

2. Loading Conditions

Same as pump requirements

3. Codes

ASME Section VIII, TEMA Class R

ASME Section III, Class III

TABLE 3.9-4 (Cont'd)

2. COMPONENT COOLING SYSTEM (Cont'd)

C. PIPING

1. Design

Piping Material	-	Carbon steel, ASTM A106, Gr B Seamless
Design pressure, psig	-	150
Design temperature, °F	-	200

2. Loading Conditions

Internal pressure and dead weight

Seismic

D. VALVES[1]

1. Design

2-1/2 inches and larger

a. Gate and globe	-	Carbon steel, butt weld ends, ANSI 150 psi
b. Check and butterfly	-	Carbon steel, flanged, ANSI 150 psi
2 inches and smaller	-	Carbon steel, socket weld ends, ANSI 150 to 600 psi

2. Loading Conditions
Internal pressure and dead weight
Seismic loadings

3. Codes

- ANSI B31.1 and ANSI B31.7, Class III, Penetration Piping is designed and fabricated to ANSI B31.7, Class I and II
- Nuclear Pump and Valve Code Class I, II and III as applicable

TABLE 3.9-4 (Cont'd)

3. CONTAINMENT SPRAY SYSTEM

A. PUMPS: 2 Required

1. Design

   TDH, ft - 470

   Temperature, F

   Maximum - 300
   Minimum - 40
   Transient 40 to 250 in 10 seconds

   Capacity, gpm - 2750 (including 50 gpm minimum recirculation)[1]

2. Fluid pumped - borated water concentration - 1900 ppm

3. Material - 304 or 316 stainless steel (casing and impeller)

4. Loading Conditions

   Located inside a seismic Class I structure; no wind loadings

   Internal pressure and deadweight
   Thermal loadings
   Nozzle loadings from connected piping
   Seismic loadings

5. Codes

   ASME Nuclear Pump and Valve Code

   Standards of Hydraulic Institute

B. SPRAY NOZZLES: 178 per spray header

1. Pressure, psig

   80 to 85 at nozzle
   40 maximum back pressure

2. Fluid - Demineralized water with 1900 to 2150 ppm boron

3. Material - 304 stainless steel

4. Loading Conditions - same as above except for nozzle loads from connected piping

5. Codes

   ASME, ASTM, ANSI

[1] As a result of modifications to the containment spray system, the system design point is different than the original design point, as revised in Chapter 6, and the minimum recirculation flow rate is increased to > 150 gpm. This has no impact on the original pump design.

TABLE 3.9-4 (Cont'd)

CONTAINMENT SPRAY SYSTEM (Cont'd)

C. VALVES[1]

1. Design

   Temperature, °F - 300

   Pressure, psig
   - 60 suction side
   - 500/550 discharge side

   Material - 304 or 316 stainless steel

   2-1/2 inches and larger
   - 150 lb design suction side
   - 300 lb design discharge side

   2 inches and smaller
   - 150 lb to 600 lb design

2. Loading Conditions

   Internal pressure and dead weight

   Seismic loadings

   Thermal loadings

3. Code – Draft ASME Code for Pumps and Valves for Nuclear Power (dated November 1968)

D. PIPING

1. Design

   Pressure, psig
   - 60 suction
   - 500 discharge
   - (some sections upgraded to 550 psig per PC/M 01059)

   Temperature, °F - 300

   Material - 304 or 316 stainless steel

   Suction Side
   - 2 inches and smaller - Schedule 40 S wall
   - 2-1/2 inches - 12 inches - Schedule 10 S or higher wall
   - 14 inches - 24 inches - 0.250 nominal wall

   Discharge Side
   - 6 inches and smaller - Schedule 40 S or higher wall
   - 8-12 inches - 0.250 or higher inches nominal wall

2. Loading Conditions

   Same as for valves

3. Code - ANSI B31.7, Class II, 1969

TABLE 3.9-4 (Cont'd)

4. INTAKE COOLING WATER SYSTEM

A. PUMPS: 3 Required

1. Design

Capacity, gpm - 14,500
TDH, feet - 130
Fluid Temperature, °F - 40 to 95
Fluid pumped - Seawater - Specific gravity 1.03
Shutoff head, feet - 200
Material - 316 Stainless Steel - Case, impeller, and shaft

2. Loading Conditions

Tornado winds of 360 mph and atmospheric depression of 3.00 psi in 3 seconds

Hurricane winds of 194 mph and atmospheric depression of 1.5 psi

Ocean wave runups to elevation +22 feet

Internal pressure and dead weight

Seismic Loadings

3. Codes

Standards of the Hydraulic Institute, ASME Section VIII, ASTM, ANSI, Nuclear Pump and Valve Code Class III.

B. VALVES[1]

1. Design

Pressure, psig - 100

Temperature, °F - 30 to 120

Fluid - Seawater

Material

Body - Carbon steel - ASTM A-285, Gr C or A-36 Cast steel - ASTM A-216, Gr WCB, Stainless steel
Disc - Ductile Ni-resist ASTM A-439, Type D-2 or Ni-resist ASTM A-436, Type 1
Shaft - SS ASTM A-276, Gr T 316
(2-1/2 inches and smaller are bronze-screwed or Stainless steel)

2. Loading Conditions

Same as for piping

3. Codes - ASME Draft Nuclear Pump and Valve Code

TABLE 3.9-4 (Cont'd)

INTAKE COOLING WATER SYSTEM (Cont’d)

C. PIPING

1. Design

Pressures, psig - 100

Temperatures F - 125

Material

14 in. and larger	-	3/8 in. wall with 1/8 in. cement lining and/or 20 mils epoxy lining, Stainless steel
3-12 in.	-	Carbon steel std wall pipe with 1/8 in. cement lining and/or 20 mils epoxy lining, Stainless steel
2-1/2 in. and under	-	Aluminum brass - ASTM, Stainless steel

2. Loading Conditions

Internal pressure and dead weight

Seismic loadings for seismic Class I portions only

3. Codes

ANSI B31.7, Class III and B31.1

TABLE 3.9-4 (Cont'd)

5. SAFETY INJECTION SYSTEM

A. HIGH PRESSURE SAFETY INJECTION PUMPS: Quantity 3

1. Design

   Capacity, gpm - 345 (including by pass Flow)

   TDH, ft - 2500

   Temperature, F
   minimum - 40
   maximum - 270

   Thermal Transient, F
   40 to 270 in 10 seconds
   270 to 40 in 10 seconds

   Fluid Pumped: borated water, 1900 ppm (min)

   Material - ASTM SA351-CF-3M

2. Loading Conditions

   Internal pressure and dead weight

   Nozzle loadings from connecting pipes including thermal loadings

   Transient loadings including thermal loadings

   Seismic loadings

3. Codes

   API standard 610, 1965 Centrifugal Pumps for General Refinery Service

   Standards of Hydraulic Institute

B. LOW PRESSURE SAFETY INJECTION PUMPS: Quantity 2

1. Design

   Capacity, gpm - 3050 (including by pass Flow)

   TDH, ft - 350

   Temperature, F
   minimum - 40
   maximum - 350

   Thermal Transient, F
   40 to 300 in 1 minute
   300 to 40 in several hours

   Fluid pumped: borated water, 1900 ppm (min)

   Material - ASTM SA351-CF3M

TABLE 3.9-4 (Cont'd)

5. SAFETY INJECTION SYSTEM

2. Loading Conditions

   Same as high pressure safety injection pumps

3. Codes

   Same as high pressure safety injection pumps

C. CHECK VALVES[1]: Quantity 2 (LP safety injection suction)

1. Design

Tag No.	-	V07000 V07001
Temperature, °F	-	350
Pressure, psig	-	300
Material	-	304 stainless steel
Type	-	300 lb ANSI

2. Loading Conditions

   Internal pressure and dead weight

   Transient loadings including thermal loading

   Seismic loadings

3. Code

   Seismic Class I

   NPVC Class II

D. SAFETY RELIEF VALVES[1]: Quantity 2 (LP safety injection suction)

1. Design

Tag No.	-	SR-07-1A
	-	SR-07-1B
Pressure, psig	-	60
Temperature, °F	-	350

   Material - 316 stainless steel

2. Loading Conditions

   Same as high pressure safety injection pumps

3. Code

   Seismic Class I

   NPVC Class II

TABLE 3.9-4 (Cont'd)

SAFETY INJECTION SYSTEM (Cont'd)

E. SHUTDOWN HEAT EXCHANGERS: Quantity 2,

1. Design

Pressure, psig - 550/150

Temperature, F - 400/250

Material shell, carbon steel
tube, austenitic stainless steel

2. Loading Conditions

Same as high pressure safety injection pumps

3. Code

ASME Section III, Class C, Shell and Tube Sides

F. SAFETY INJECTION TANKS: Quantity 4

1. Design

Pressure, psig - 280

Temperature, F - 200

Material - SA264

2. Loading Conditions

Internal pressure and deadweight

Nozzle loadings from connecting pipes including

thermal loadings

Seismic loadings

3. Code

ASME Section III, Class C

6. WASTE MANAGEMENT SYSTEM

A. TANKS

1. Design

a. Gas Surge
Pressure, psig - 40
Temperature, F - 200

TABLE 3.9-4 (Cont'd)

WASTE MANAGEMENT SYSTEM (Cont’d)

b. Gas Decay
Pressure, psig - 190
Temperature, F - 250

c. Spent Resin
Pressure, psig - 50
Temperature, F - 200

d. Flash
Pressure, psig- 70 internal, 15 external
Temperature, F- 250

2. Loading Conditions

Internal pressure and dead weight
Nozzle loadings from connecting pipes including thermal loadings
Seismic loadings
Same as purification filter for flash tank

3. Code ASME Section III, Class C

B. REACTOR DRAIN PUMPS Quantity 2

1. Design

Capacity, gpm - 50
Head, ft - 140
Temperature, F
minimum - 70
maximum - 150

2. Code

Draft ASME Code for Pumps & Valves for Nuclear Power,
Nov. 1968, Class III, Issued for Trial Use & Comment

C. REACTOR DRAIN TANK

1. Design

Pressure, psig - 25 internal, 15 external
Temperature, F - 250

2. Code*

ASME Section III, Class C

TABLE 3.9-4 (Cont'd)

WASTE MANAGEMENT SYSTEM (Cont'd)

D. HOLDUP TANK

1. Design

   Pressure, psig - 10 internal, 2 external
   Temperature, F - 240

2. Code

   ASME Section VIII, Div. 1

E. ION EXCHANGERS (PRECONCENTRATOR AND BORIC ACID CONDENSATE)

1. Design

   Pressure, psig - 150
   Temperature, - 250

2. Code

   ASME Section VIII, Div. 1

F. PRECONCENTRATOR FILTER

1. Design

   Pressure, psig - 150
   Temperature, F - 240

2. Code

   ASME Section III, Class 'C'

7. FUEL POOL SYSTEM

A. HEAT EXCHANGER: Quantity 1

1. Design

   Pressure, psig - 150/75

   Temperature, F - 250

   Materials shell side - carbon steel
   tube side - austenitic stainless steel

2. Loading Conditions

   Internal pressure and dead weight

   Nozzle loadings from connecting pipes including thermal loading

   Seismic loadings

3. Code ASME Section III, Class C, Shell and Tube Sides

TABLE 3.9-4 (Cont'd)

FUEL POOL SYSTEM (Cont'd)

B. ION EXCHANGER: Quantity 1

1. Design

   Pressure, psig - 200
   Temperature, F - 250
   Material - 304 stainless steel

2. Loading Conditions

   Same as heat exchanger

3. Code

   ASME Section VIII, Div. 1

C. FILTER: Quantity 1

1. Design

   Pressure, psig - 150

   Temperature, F - 200

2. Loading Conditions

   Same as heat exchanger

3. Code

   ASME Section VIII, Div. 1

D. PIPING

ASME Section B31.1

E. PUMPS: Quantity 3

Fuel Pool (2) - Draft ASME Code for Pumps & Valves for Nuclear Power, Nov. 1968, Class III, Issued for Trial Use & Comment

Purification (1) - Draft ASME Code for Pumps & Valves for Nuclear Power, Nov. 1968, Class III, Issued for Trial Use & Comment

TABLE 3.9-4 (Cont'd)

8. CHEMICAL AND VOLUME CONTROL SYSTEM

A. REGENERATIVE HEAT EXCHANGER Quantity 1

1. Design

   Pressure, psig - 3025/2485
   Temperature, F - 650
   Material - Tube side - austenitic stainless steel
   Shell side - austenitic stainless-steel

2. Loading Conditions

   Internal pressure and dead weight
   Nozzle loadings from connecting pipes including thermal loadings
   Transient loadings including thermal loadings Seismic loadings

3. Code

   ASME Section III, Class C

B. LETDOWN HEAT EXCHANGER Quantity 1

1. Design

   Pressure, psig - 650/150
   Temperature, F - 550/250
   Material - Tube side - austenitic stainless steel
   Shell side - carbon steel

2. Loading Conditions

   Same as regenerative heat exchanger

3. Code

   ASME Section III, Class C

C. ION EXCHANGERS (PURIFICATION AND DEBORATING) Quantity 3

1. Design

   Pressure, psig - 200
   Temperature, F - 250
   Material - 304 stainless steel

2. Loading Conditions

   Internal pressure and dead weight
   Nozzle loadings from connecting pipes including thermal loadings
   Seismic loadings

TABLE 3.9-4 (Cont'd)

CHEMICAL AND VOLUME CONTROL SYSTEM (Cont'd)

3. Code

   ASME Section III, Class C

D. PURIFICATION FILTERS Quantity 2

1. Design

   Pressure, psig - 200
   Temperature, F - 250

2. Loading Conditions

   Internal pressure and dead weight
   Nozzle loadings from connecting pipes including thermal loadings
   Seismic loadings

3. Code*

   ASME Section III, Class C

E. TANKS (VOLUME CONTROL) Quantity 1

1. Design

   Pressure, psig - 75 internal, 15 external Temperature, F - 250

2. Loading Conditions

   Same as purification filter

3. Code*

   ASME Section III, Class C

F. BORIC ACID MAKE UP TANKS

1. Design

   Pressure, psig - 15 internal, 0 external
   Temperature, F - 200
   Material - SA 240 TP 304

* Original procurement information

2. Loading Conditions

   Internal pressure and dead weight
   Nozzle loadings from connecting pipes including thermal loadings
   Transient loadings including thermal loadings
   Seismic loadings

3. Code

   ASME Section III, Class C

G. <u>BORIC ACID MAKEUP PUMPS</u> Quantity 2

1. Design

   Pressure, psig - 150
   Temperature, °F - 250

2. Loading Conditions - same as F.

3. Code

Draft ASME Code for Pumps & Valves for Nuclear Power, Nov. 1968, Class II, Issued for Trial Use & Comment

Notes:

1. Material designation and pressure classes will be changing for safety related valves (2 inches and smaller) in accordance with PC/M 92282.

TABLE 3.9-5

ACTIVE PUMPS AND VALVES OUTSIDE
REACTOR COOLANT PRESSURE BOUNDARY

Pumps

High Pressure Safety Injection (2)*

Low Pressure Safety Injection (2)

Containment Spray (2)

Charging (3)

Boric Acid Makeup (2)

Component Cooling (3)

Intake Cooling (3)

Auxiliary Feedwater (3)

Valves

High Pressure Safety Injection (8)

Low Pressure Safety Injection (4)

Containment Spray (2)

Iodine Removal Flow Control (2) (option)

Shutdown Cooling Heat Exchanger

Shutdown Cooling Heat Exchanger Flow Control

Component Cooling (6)

Intake Cooling (4)

Auxiliary Feedwater (4)

Main Steam Line (2)

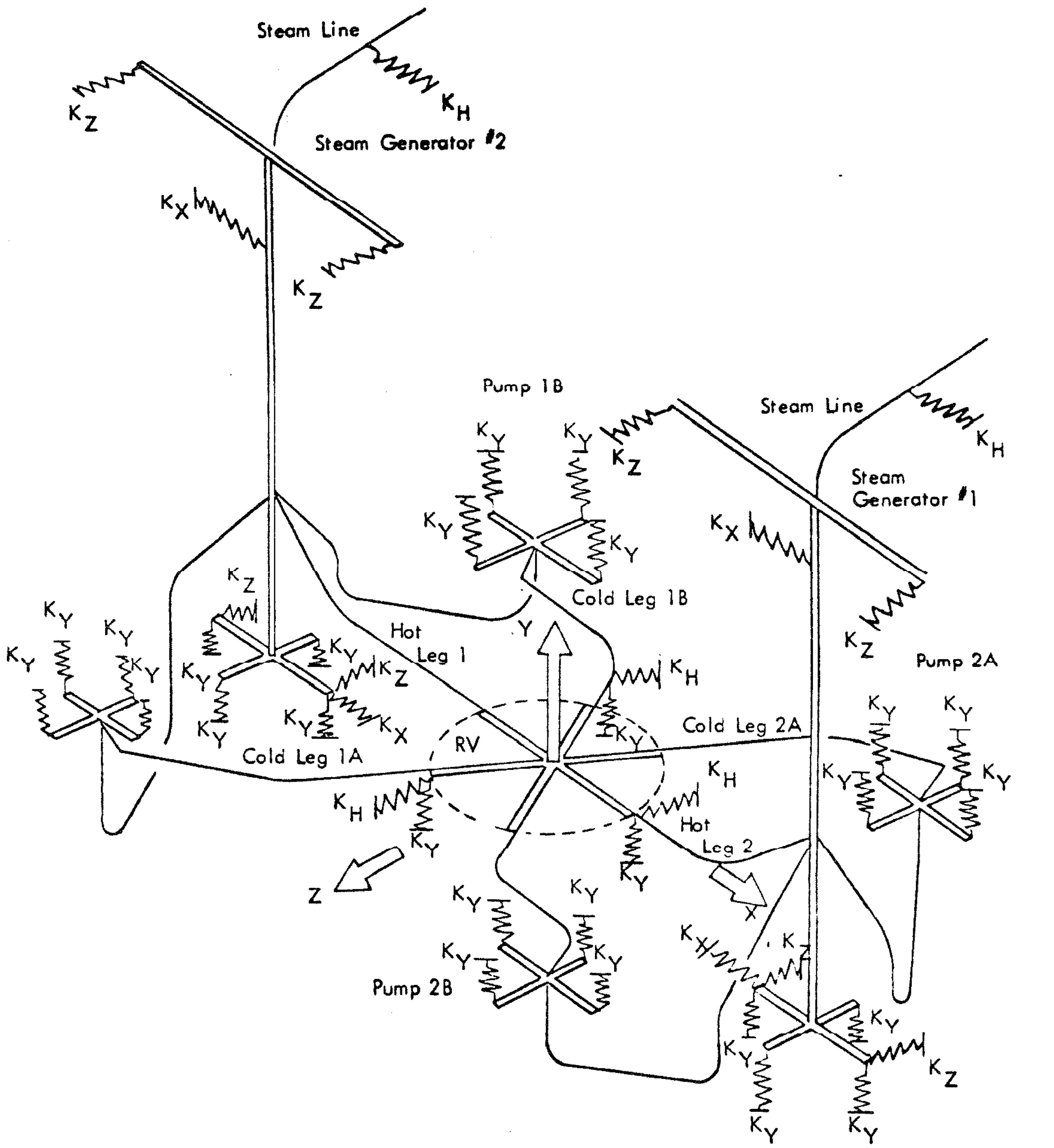
Containment Isolation

*Note: HPSI pump 1C has been abandoned in place and isolated by locked closed valves.

Table 3.9-6

ST. LUCIE UNIT 1 VALVES WITHIN THE SCOPE OF GENERIC LETTER 89-10

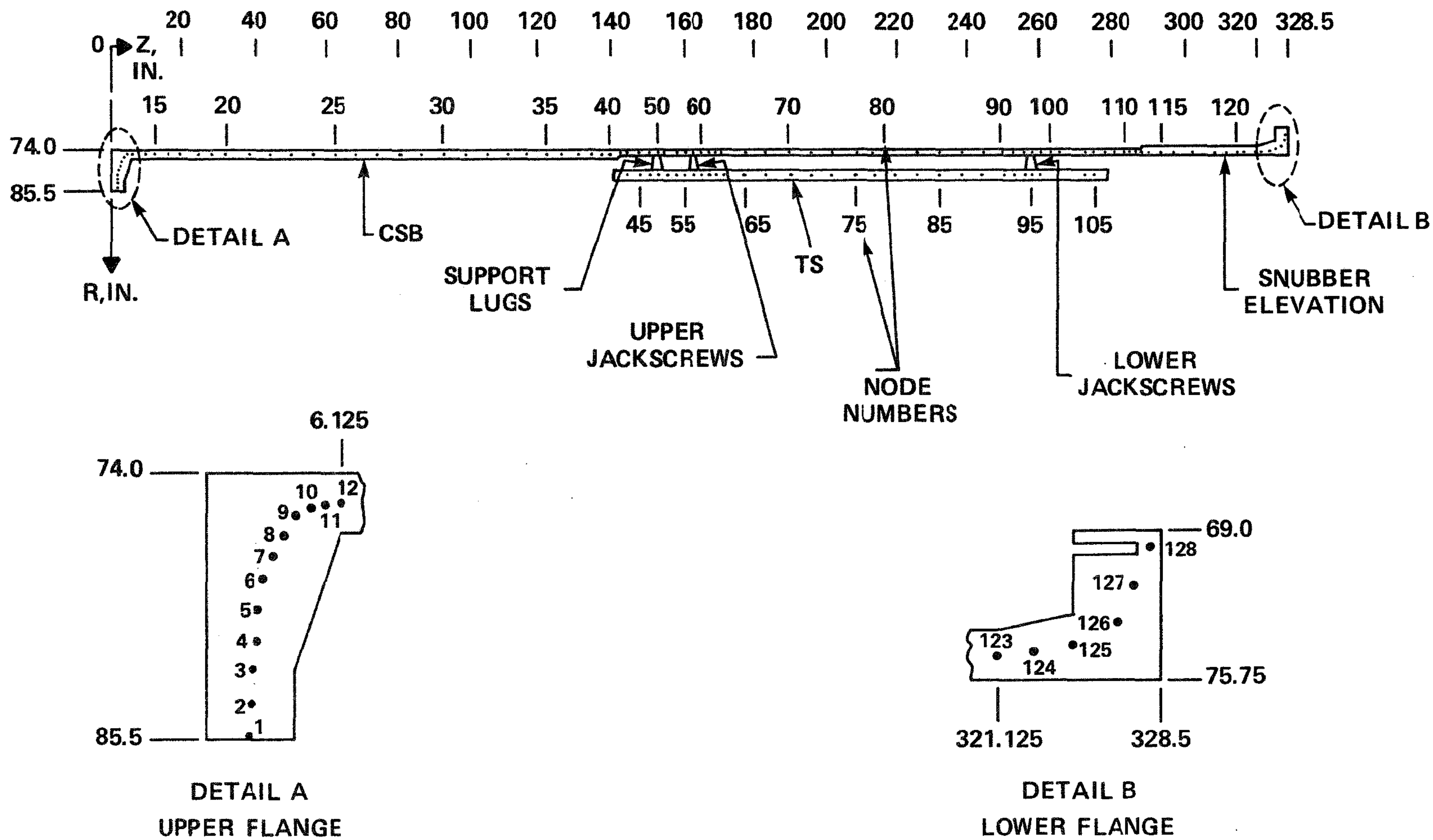
VALVE #	VALVE #
V1403	MV-03-1A
V1405	MV-03-1B
V2501	MV-03-2
V2504	MV-07-1A
V2508	MV-07-1B
V2509	MV-07-2A
V2514	MV-07-2B
	MV-08-1A
V3206	MV-08-1B
V3207	MV-08-3
V3432	MV-08-13
V3444	MV-08-14
V3452	MV-09-1
V3453	MV-09-2
V3456	
V3457	
V3480	MV-09-9
V3481	MV-09-10
HCV-3615	MV-09-11
HCV-3616	MV-09-12
HCV-3617	
HCV-3625	
HCV-3626	
HCV-3627	
HCV-3635	
HCV-3636	
HCV-3637	
HCV-3645	
HCV-3646	
HCV-3647	
V3651	MV-15-1
V3652	MV-18-1
V3654	MV-21-2
V3656	MV-21-3
V3659	
V3660	
V3662	
V3663	



TRANSLATIONAL FLEXIBILITY

SUPPORT	K_X (in/lb)	$K_Y^{(DN)}$ (in/lb)	$K_Y^{(UP)}$ (in/lb)	K_Z (in/lb)	K_H (in/lb)
Reactor Vessel	--	1×10^{-14}	1	--	1×10^{-14}
Coolant Pump	1	1×10^{-4}	1	1	1
Lower St Gen	1×10^{-12}	1×10^{-12}	5.38×10^{-8}	2×10^{-10}	--
Upper St Gen	1×10^{-7}	1	1	3.96×10^{-9}	--
Steam Line	--	1	1	--	1×10^{-6}

FLORIDA POWER & LIGHT CO. St. Lucie Plant	Reactor Coolant System Supports Mathematical Model	Figure 3.9-1



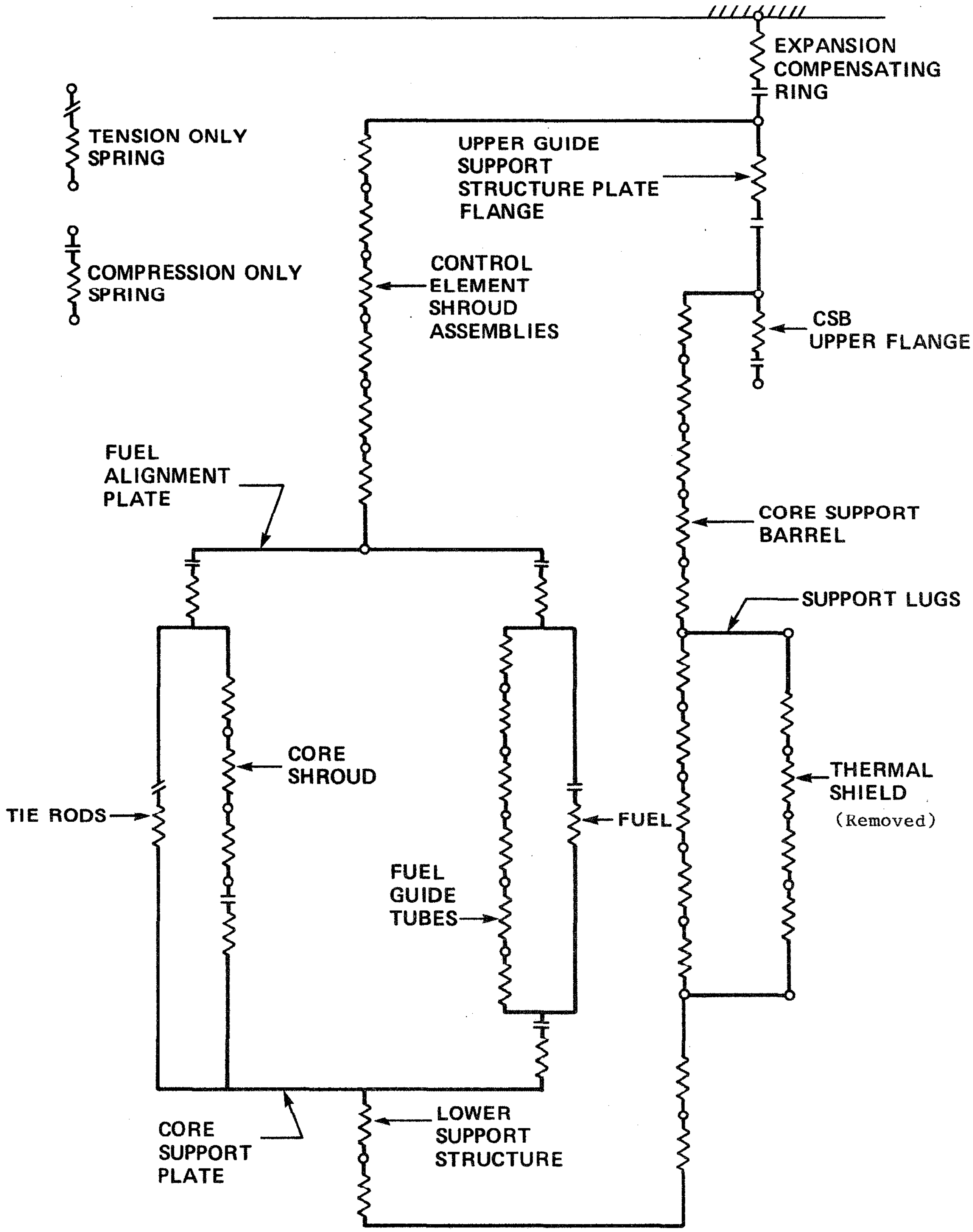
FOR HISTORICAL INFORMATION ONLY

Am. 3-7/85

FLORIDA POWER & LIGHT CO.
St. Lucie Plant
Unit 1

ASHSD FINITE ELEMENT MODEL OF THE CSB/TS SYSTEM

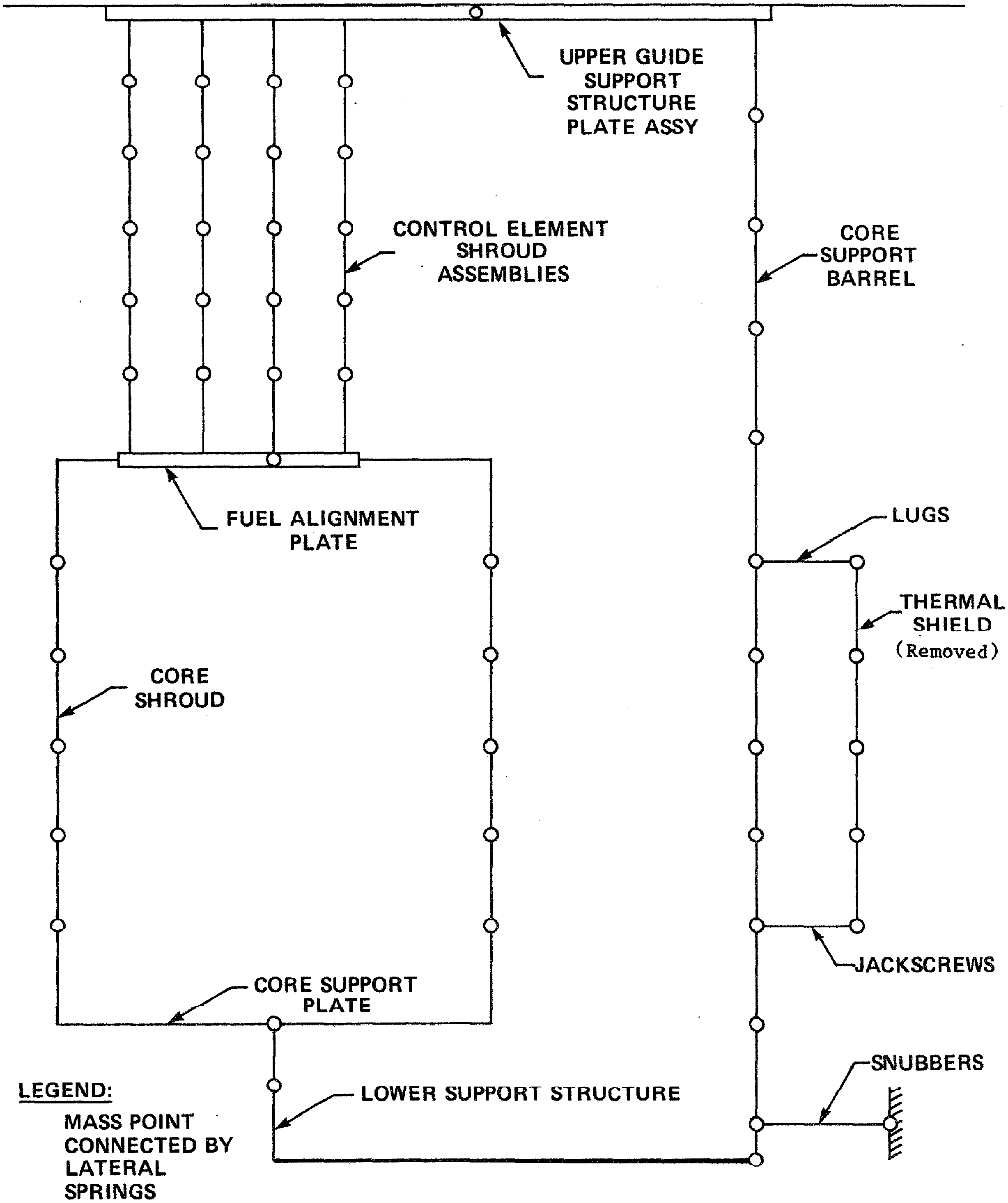
Figure 3.9-2



Am. 3-7/85

FOR HISTORICAL INFORMATION ONLY

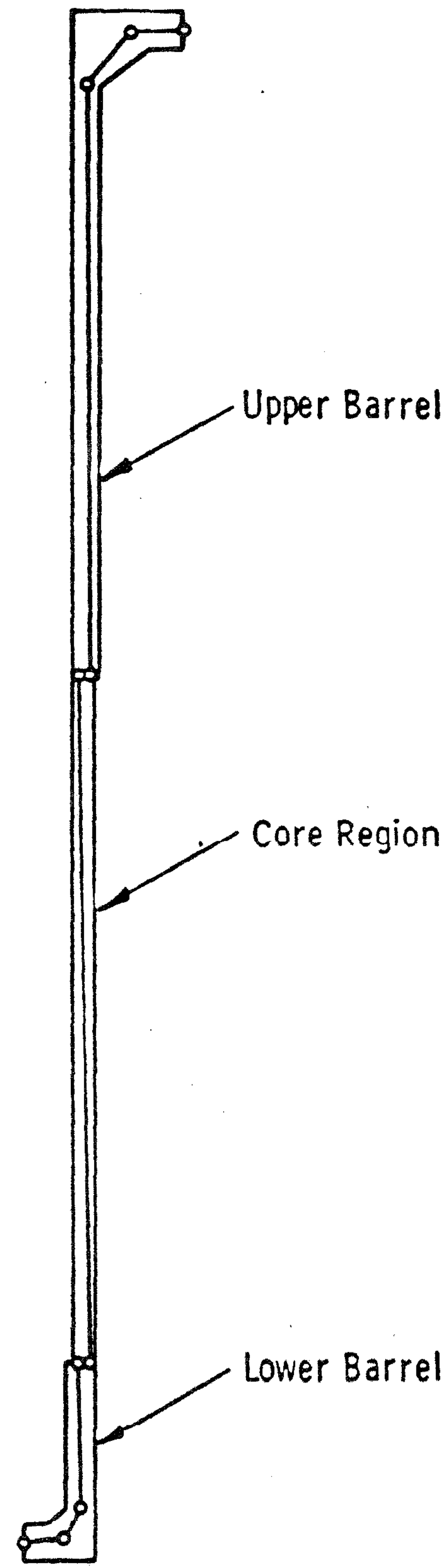
FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	VERTICAL SHOCK MODEL	Figure 3.9-3



Am. 3-7/85

FOR HISTORICAL INFORMATION ONLY

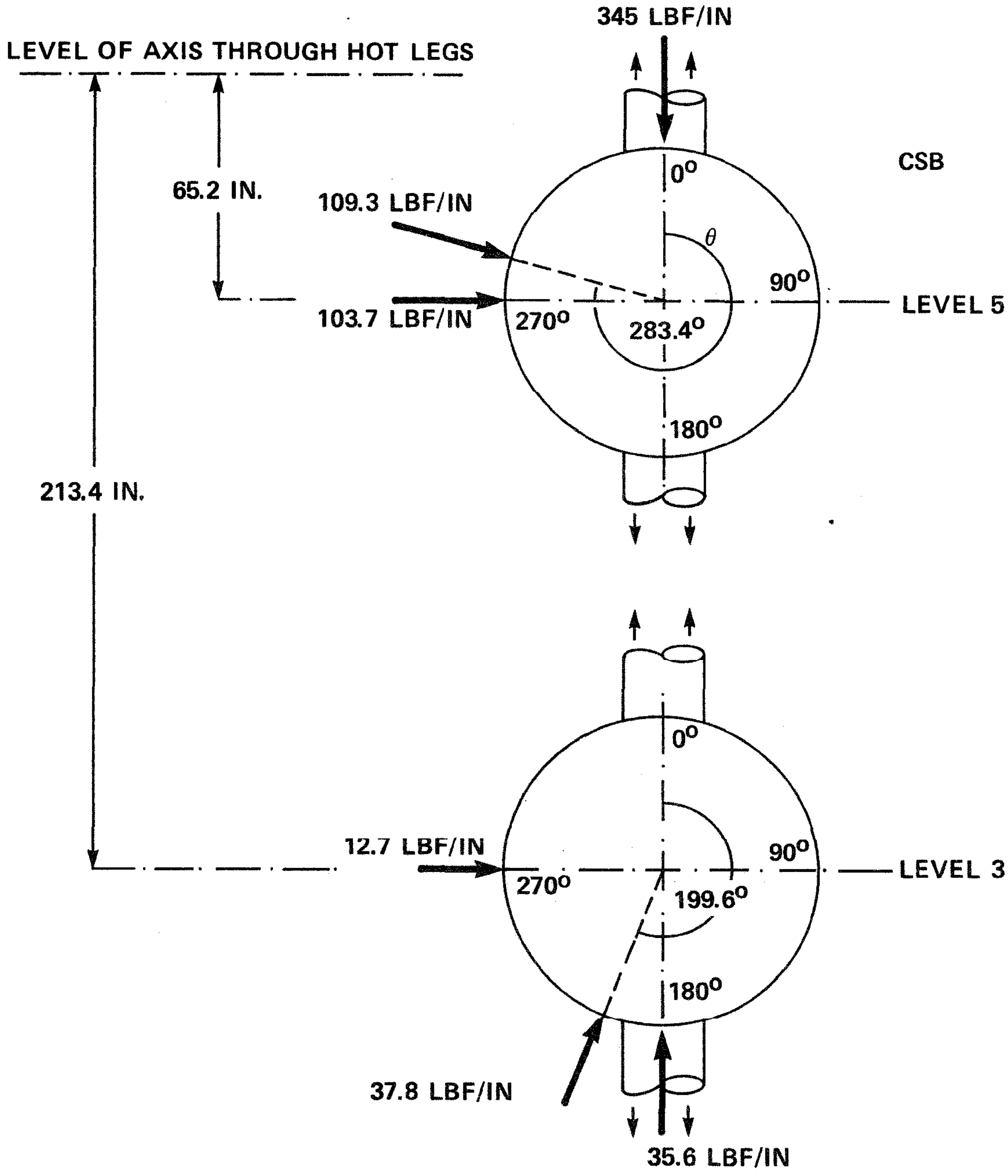
FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LATERAL SHOCK MODEL	Figure 3.9-4



FOR HISTORICAL INFORMATION

Am. 3-7/85

FLORIDA POWER & LIGHT CO. St. Lucie Plant	SAMMSOR/DYNASOR Finite Element Model of Core Support Barrel	Figure 3.9-5



T_{IN} = 500 °F; Q = 130 % OF QDES

Am. 3-7/85

FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	NORMAL OPERATING LATERAL HYDRAULIC LOADS ON THE CORE SUPPORT BARREL	Figure 3.9-6

3.10 SEISMIC DESIGN OF CLASS I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

3.10.1 SEISMIC CRITERIA

Instrumentation and electrical equipment associated with the reactor protective system, engineered safety features actuation system (ESFAS) and on-site emergency power system is designed as seismic Class I to ensure their ability to initiate protective action during, and following a design basis earthquake (DBE) and to supply standby electrical power following a DBE to components required to mitigate the consequences of a LOCA.

Safety related cable tray support systems are designed to the following seismic criteria:

a) Class I electrical cable tray support structural integrity shall not be violated during a design basis earthquake

b) Cable tray support design shall be such that the natural frequency of the loaded tray support system is not in resonance with the seismic forcing function

c) Analysis shall verify that loaded trays between the maximum support span can withstand seismic accelerations

d) Acceleration response spectra utilized in the analysis shall represent the response spectra for the floor elevation to which the supports are welded

e) All safety related cable tray supports shall be fabricated by a vendor who has been qualified to manufacture material for seismic Class I service.

3.10.2 SEISMIC QUALIFICATION OF COMPONENTS

Purchase specifications for seismic Class I instrumentation and electrical equipment contain horizontal and vertical seismic acceleration values based on the DBE spectra for the equipment location.

Where there is a possibility of amplification or the floor response acceleration due to equipment supports, the specified acceleration values are increased to account for such amplification.

The vendors are also supplied the floor response spectra for the equipment location. The vendors are required to ensure that the natural period of vibration of the component does not fall within the critical frequency range of the floor response spectra. The vendors are required to submit qualification data which demonstrate that the component is capable of functioning under the specified seismic loadings. The qualification data may consist of prototype test results, mathematical analysis or operational experience.

Table 3.10-1 lists seismic qualification data for Class I instrumentation and electrical components of the reactor protective system, ESFAS and emergency power system. The table lists the specified seismic acceleration values, the type of qualification performed and the acceleration values for which each component was qualified. Where tests were performed, the testing generally consisted of shock table vibration tests. Such tests were performed over a range of frequencies corresponding to the expected range of seismic frequencies associated with the DBE.

Vendors were required to submit documentation of the seismic qualification of each component. This documentation consists of certification that the component is capable of functioning during and following the specified seismic loading together with the mathematical calculations or test results.

The seismic qualification provisions generally meet the requirements of IEEE-344-1971, "IEEE Guide for Seismic Qualification of Class IE 1E Electric Equipment for Nuclear Power Generating Stations," although the guide was not in existence at the time of issuance of the Hutchinson Island plant construction permit and is therefore not included as a specific requirement in the purchase specifications.

References 1 through 7 contain vibration data for electrical and instrumentation equipment that have been seismically tested and are directly applicable to the St. Lucie Plant (formerly Hutchinson Island). The floor response spectra for each of the buildings housing seismic class I instrumentation and electrical equipment, shown on Figures 3.7-12 through 3.7-24, falls within the test response spectra envelope presented in Figure 7 of Reference 1 indicating that the test conditions were more severe than that would be experienced during a postulated DBE.

REFERENCES FOR SECTION 3.10

1) CENPD-61, "Seismic Qualification of Category I Electric Equipment for Nuclear Steam Supply Systems," November 1972.

2) CENPD-61 (Supplement 1) Same title, not dated

3) CENPD-61 (Supplement 2) Same title, February 19, 1973

4) CENPD-61 (Supplement 3) Same Title , March 2, 1973

5) CENPD-61 (Supplement 3) Rev. 1, March 2, 1973

6) CENPD-61 (Supplement 4) Same Title, Not dated

7) CENPD-61 (Supplement 5) Same Title, August 23, 1973

TABLE 3.10-1

SEISMIC QUALIFICATION DATA FOR CLASS I INSTRUMENTATION AND ELECTRICAL EQUIPMENT

Component	Specified Accelerations Horizontal (g)	Specified Accelerations Vertical (g)	Vendor's Seismic Qualifications Procedures	Accelerations Used in Vendor's Program	NOTES
Cable Tray Supports:					
a) Reactor Auxiliary Building	0.52	0.34			
b) Reactor Building	0.3	0.2			
Emergency Power System:					
a) Diesel generator sets	0.3	0.2	Calculations	Specified values	4
b) Diesel generator control panels	0.3	0.2	Test	Specified values	
c) 4.16 kv switchgear	0.3	0.1	Test	Specified values	3
d) 480 v switchgear	0.3	0.1	Test	Specified values	3
e) 480 v MCCs	0.3	0.1	Test	0.5 g over a frequency of 5 - 500Hz	3
f) 125 v ac and dc panels	0.3	0.1	Test	Specified values	3
g) Station battery cells	0.41	0.34	Test	Per Wyle Laboratories Test Program No.43450-1	3
h) Battery Racks	0.41	0.34	Test	Per Wyle Laboratories Test Program No.43450-1	4
i) Battery chargers	0.3	0.1	Test	1.96 g (hor); 1.28 g (vert)	3
j) Instrument Inverters	0.3	0.1	Test	1.96 g (hor); 0.5 g (vert)	3, 5
k) Penetration assemblies	0.23	0.15	Test	Specified values	3
Penetration assemblies (C3, C6, D10)	0.23	0.15	Calculation	Specified values	4
l) Station service transformers	0.3	0.1	Test	Per Wyle Seismic Report 48916-1	3

TABLE 3.10-1 (Cont'd)

Component	Specified Accelerations Horizontal (g)	Specified Accelerations Vertical (g)	Vendor's Seismic Qualifications Procedures	Accelerations Used in Vendor's Program	NOTES
Engineered Safety Features					
Actuation System (ESFAS):					
a) Pressurizer pressure sensors			Test		1
b) Steam generator pressure sensors			Test		1
c) Containment pressure sensors	0.4	0.27	Test	1.0 g (hor); 1.0 g (vert)	
d) Containment radiation sensors	1.5	1.0	Test	Specified values over a frequency of 5 - 60 Hz	
e) Refueling water tank level sensors	0.5	0.33	Test	1.0 g (hor); 1.0 g (vert)	
f) Measurement cabinets	0.8	0.6	Test & Calculation	Specified values	3,4
g) Actuation cabinets	0.8	0.6	Test & Calculation	Specified values	3,4
Reactor Protective System					1
Reactor Turbine Generator Board	0.3	0.3	Prototype Test	Specified values	2

NOTES:
1) Seismic qualification data included in C-E report CENPD 61 referenced in Section 1.6.
2) Includes all safety related devices and annunciators mounted on the board.
3) Refer to Appendix 3B for a summary of test results for original equipment.
4) Calculations reviewed as discussed in Section 3.7.5.
5) Wyle Test Report 53116-1 (3/14/06) (Inverters 1A and 1C only)

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT

St. Lucie Unit 1 was originally required to meet IEEE 323-1971 for the environmental qualification of equipment. In January 1980 the NRC issued IE Bulletin 79-01B to which FPL responded. In February 1983, Congress codified the requirements for the environmental qualification of electrical equipment in Chapter 10, Part 50, Section 49 of the Code of Federal Regulations (10 CFR 50.49). This section required all holders of an operating license issued prior to February 22, 1983 to develop and complete a program for the qualification of equipment subject to 10 CFR 50.49 by the end of the second refueling outage after March 31, 1982 or by March 31, 1985 whichever came first.

Pursuant to the requirements of 10 CFR 50.49, FPL has established a program for qualifying the electrical equipment defined in paragraph (b) of 50.49. The list of equipment subject to these requirements is provided in drawing 8770-A-450, Environmental Qualification (EQ) List for 10 CFR 50.49 Environmental Qualification Documentation Packages (Doc Pacs, drawing series 8770-A-451) provide the qualification documentation for this equipment. Doc Pac 1000 (drawing 8770-A-451-1000) provides the "design basis" for the program.

The remainder of Section 3.11 provides the original environmental design of mechanical and electrical equipment. In cases where this section overlaps with 10 CFR 50.49, the EQ List and Doc Pacs described above supersede the information provided here. Therefore, the EQ List and appropriate Doc Pacs should be consulted prior to use of the information in the remainder of this section.

Pursuant to the requirements of IE Bulletin 79-01B FPL reevaluated the environmental qualification of equipment used for accident mitigation or post-accident monitoring irrespective of whether it is categorized as Class 1E or not. This equipment was evaluated against the Enclosure 4 Guidelines of the bulletin as discussed in the phase II report responding to the bulletin. The evaluation included a detailed review of the design basis accident service conditions during which accident mitigating and post-accident monitoring equipment is required to function and qualification test reports were reviewed to ensure the equipment was tested within those service conditions.

The service conditions considered were:

1) Inside Containment for a Loss of Coolant Accident

   Application of the Enclosure 4 Guidelines in this case reveals that the environmental conditions of temperature, pressure, humidity for the long-term (180 days), radiation for up to one year and short-term (up to 15 minutes) given in Section 3.11.1 are applicable.

   Clarification of these guidelines were provided as follows:

   a) For beta radiation dose a demonstration was provided, as part of the auditible records, showing that the total beta dose is less than 10 percent of the total gamma dose. This allows equipment which is qualified for the total radiation environment (gamma plus beta).

3.11 ENVIRONMENTAL DESIGN OF MECHANICAL AND ELECTRICAL EQUIPMENT (Cont'd)

b) Refinements were made in some cases to the equipment operability times and resulting service conditions at these times. Certain items, such as containment isolation valves and SIAS/CIAS transmitters, are required to function for only a short period of time at the onset of the accident and having performed their function are no longer required. Thus the service conditions used for qualification were well below the maximum DBA-LOCA conditions of 36.1 psig and 256.6°F.

c) The flood level considered for submergence of equipment (27.3 feet) is that level resulting from injection of the entire useable volume of the Refueling Water Tank into containment. The actual maximum level would be approximately 2 feet lower.

d) The pH of the containment spray is maintained between about 7.89 and 9.49 during both injection and recirculation by the addition of sodium hydroxide. This chemical spray environment was considered to exist for only a limited period following which the equipment begins to dry.

2) Inside Containment for Main Steam Line Break

Per Enclosure 4 Guidelines, the Chapter 15 analyses are adequate to determine the steam/air pressure and temperature service conditions for a DBA-MSLB.

Clarification of these guidelines were provided as follows:

a) For the worst case MSLB accident the containment atmospheric temperature exceeds that given in Table 3.11-1 for approximately 40 seconds. This temperature (290°F) was not used as a basis for environmental qualification of electrical equipment due to the extremely short time of the transient as discussed in Subsection 6.2.1.3.

b) MSLB radiation, submergence and spray qualification parameters are enveloped by or analogous to those described for a LOCA above.

3) Service Conditions Outside Containment

Per Enclosure 4 Guidelines this service condition was split into 3 parts:

a) Areas Subject to a Severe Environment as a Result of a High Energy Line Break

Enclosure 4 states that electrical equipment located in these areas should be qualified for the service conditions reviewed and approved in the HELB Safety Evaluation Report for each specific plant.

This was interpreted to allow the use of the accident environments specified in Appendices 3C and 3D for ruptures outside containment of a main steam line, a main feedwater line, a steam generator blowdown line, an auxiliary steam line, a CVCS letdown line, and a CVCS charging line. The appendices did not discuss a rupture in the auxiliary feedwater system. This was added and the environment specified as follows. Up to 120°F; 14.7 psia since an outside area; up to 100% relative humidity only if the break is directed towards equipment under consideration.

b) Area Where Fluids are Recirculated from Inside Containment to Accomplish Long-Term Cooling Following a LOCA

Enclosure 4 states that temperature and pressure should be based on plant unique analysis, that one hundred percent humidity should be considered in confined spaces, and that radiation service conditions must be evaluated on a case by case basis.

The guidelines were clarified to limit the area to the ECCS area. ECCS leakage was expected to be minimal and HVAC systems were assumed to operate, thus pressure/temperature/humidity values remain near ambient conditions. Equipment radiation dose maps were generated which account for radiation due to circulation of post LOCA sump fluid. The source terms were based on NUREG-0578 guidance.

c) Areas Normally Maintained at Room Conditions

Enclosure 4 states that equipment in these areas does not experience significant stress due to a change in service conditions during a design basis event and that failures are expected to be random.

No accident mitigating or post-accident monitoring equipment was reviewed for these areas per guidance provided in Question/Answer No. 1 of the IE Bulletin 79-01B Supplemental Information dated 2/29/80, and in the NRC Regional Meetings held July 14-17, 1980.

The results of this evaluation were reported in References 2, 3, 4 and 6. Certain components failed to meet this criteria. However, justification for continued operation along with an estimated schedule for resolution was provided.

The information presented in the references was accurate at the time of submittal, however, changes have subsequently been made and reported to the NRC. See Reference 5.

The environmental qualification criteria adopted as discussed above is now governing for all electrical equipment which has an accident mitigating or post-accident monitoring function and may be exposed to harsh environments during and after an accident. Therefore, the information in the text that follows does not stand alone and must be reviewed in conjunction with the 79-01B report found in References 2 through 6.

3.11.1 EQUIPMENT IDENTIFICATION

3.11.1.1 Environmental Design Criteria

The design criteria with respect to environmental effects on the electrical and mechanical equipment of the reactor protective system and engineered safety features to assure acceptable performance in all environments (normal and accident) are based on equipment location. As far as practical, equipment for these systems is located outside the containment or other areas where high activity levels or adverse environmental conditions could exist.

The reactor protective system and engineered safety features are capable of performing their intended functions under the following specified environmental service conditions:

a) All reactor protective system and engineered safety features components are capable of meeting their rated performance specifications under the environmental service conditions expected as a result of normal operating requirements and expected extremes in operating requirements.

b) All reactor protective system and engineered safety features equipment required to accomplish protective actions in response to a design basis event are capable of completing their function under the environmental service conditions related to the design basis event. The environmental service conditions related to a design basis event are specified to include normal operating conditions before the event, conditions produced by the event, and conditions existing subsequent to the event for such time as is required for the protective actions to be carried to completion.

c) The reactor protective system and engineered safety features equipment are capable of meeting the specified performance requirements for the most degraded conditions resulting from the long term environment to which the equipment is normally exposed.

d) All reactor protective system and engineered safety features equipment are qualified for use under the specified environmental service conditions in accordance with IEEE-323-1971, "IEEE Trial-Use Standard: General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations."

3.11.1.2 Environmental Service Conditions

The plant environmental service conditions are classified in the following environmental design categories.

a) I-A - Long term containment environment following LOCA or steam line break accident

b) I-B - Short term containment environment following LOCA or steam line break accident

c) I-C - Containment environment following all other design basis accidents

d) I-D - Control room environment following loss of air conditioning

3.11.1.3 Equipment Classification

Reactor protective system and engineered safety feature components are classified according to the environmental design categories depending upon their location and functional requirements. Functional requirements are determined by assessing the impact of a safety related component's failure on the ability to mitigate the effects of a LOCA or to shutdown. On that basis the following equipment is included in each category.

a) Category I-A

Containment fan coolers
Containment - annulus ΔP sensor
Containment electrical penetration assemblies
H_2 Sampling Solenoids I-FSE-27-1, 2,3,4,5,6,7
Shutdown Cooling System Isolation Valves: V3480, V3481,
V3651, and V3652
Electrical cables associated with above

b) Category I-B

Pressurizer pressure sensors
Containment pressure sensors
Steam Generator pressure sensors
Containment radiation sensors
Containment isolation valves
Electrical cables associated with above
SIT isolation valves[(1)]: V3614, V3624, V3634, and V3644

c) Category I-C

Reactor coolant temperature sensors
Reactor coolant flow ΔP sensors
Out of core neutron detectors
Steam generator level transmitters
Electrical cables associated with above

d) Category I-D

Reactor protective system cabinets
ESFAS measurement and actuation cabinets
Reactor protective system trip switchgear
Electrical cables associated with above

---

(1) Technical specifications require valves be open with power locked out obviating the need for the qualification test performed.

The components listed in each category are designed to operate in the temperature, pressure, humidity, and radiation environment for the time shown on Table 3.11-1 unless detailed calculations are provided to justify less stringent environmental design criteria. Components in categories I-A and I-B are designed to withstand the environmental conditions shown in addition to long term operation in category I-C environment. The containment radiation environment is based on a LOCA fission product release source consisting of 50 percent of the core halogen inventory 100 percent of core noble gas inventory and one percent of core solid fission product inventory.

The integrated dose used for original license for the electrical penetrations was determined by analysis as discussed below. The dose assumed 40 years of normal operation, a LOCA, and one year of post LOCA operation. The fission product source term was as cited above with a uniform dispersion of nuclides within the containment. No fission product removal credit was taken for containment sprays which would be operative during and subsequent to a LOCA. Both gamma and beta contributions to the dose were determined. For the gamma dose regions of the containment where the line of sight between source and receptor passes through significant shielding barriers may be appropriately excluded from the analyses. However, for this analysis only the 4 foot thick secondary concrete shield wall was considered an effective shielding barrier. The analyses yielded the following:

	dose (R)
40 years of normal operation	1.8×10^{3}
0 to 2 hour LOCA dose	1.3×10^{6}
2 to 24 hour LOCA dose	4.1×10^{6}
1 to 31 day LOCA dose	1.7×10^{7}
31 to 365 day LOCA dose	4.0×10^{6}
Total integrated dose	2.7×10^{7}

The Components of the start-up range (rate of power change protection) power range (over-power protection) and power range control channels as shown on Figures 7.2-4 and 7.2-5 are installed in oil-tight troughs. The 6.9 KV power cable for the reactor coolant pump motors are also routed in this tunnel in cable tray and conduit.

The environmental design criteria for nuclear instrumentation components located within the cable tunnel is that they meet environmental service condition I-C which does not require operation following a LOCA or steam line break accident.

Only cables to components that are required to function in the long term post-LOCA containment environment need be qualified for this environment. Nonetheless, all safety related and post-LOCA monitoring cables inside containment will be qualified for the long term containment environment for all accident and incident conditions (Category I-A). With regard to the electrical tunnel, the cables serving the out-of-core detectors will be Category IA cable.

The location of motor operated valves in the containment has been reviewed to determine if they could become submerged in the post LOCA environment. Table 3.11-1A lists the motor operated valves located inside containment and their elevations. The maximum post LOCA water level of approximately 24 feet would submerge valves V3614, V3624, V3634, and V3644. There are no adverse consequences associated with the submergence of these valves in the short or long term ECCS functions or containment isolation. The subject valves are normally locked open valves on the safety injection tank lines to the primary system. The safety injection tank function will be over long before the valves are submerged (approximately 30 minutes for submergence) which negates the consequences of any postulated failure mode and they serve no containment isolation function. In view of the above no modification is necessary.

As part of the Post TMI-Short Term Lessons Learned studies a design review of plant shielding and environmental qualification of equipment for spaces/systems which may be used as post accident operations was performed for PSL-1 (Item 2.1.6.b). See Reference 1. The objective of the review in part was to ensure that access to and operation of safety related equipment needed for postaccident operations would not be impaired by high radiation fields.

Radiation fields were calculated considering gaseous nuclide releases from the core into the containment atmosphere as a result of the postulated LOCA as well as fields resulting from all equipment and piping outside the containment which might contain primary coolant or radioactive gases during post-accident operation.

Source terms for liquid systems were assumed to be 100% of the core inventory of noble gases, 50% of the core inventory of halogens, and 1% of the core inventory of other nuclides contained in the primary coolant. Source terms for containment atmosphere followed the guidance of Regulatory Guide 1.4. Appropriate dilution and decay factors were applied.

Acceptance criteria for equipment doses were determined either by equipment specification, actual equipment qualification, or generic material damage. No requirements for corrective action were presented in Reference 1 rather, equipment dose data has been factored into ongoing studies of equipment environmental qualification. See Section 12.1.6 for information with respect to operator dose.

Future qualification of components for radiation inside containment will use the "Equipment Qualification Radiation Dose Map Development" submitted via Reference 7.

3.11.2 QUALIFICATION TESTS AND ANALYSES

The equipment in each environmental design category is qualified by the following methods, which are in accordance with IEEE-323 (1971).

a) Category I-A and I-B Equipment

The equipment is qualified for the specified environmental conditions by a combination of type testing and analysis. Type tests are performed wherein the components are subjected to the specified temperature, pressure and humidity conditions. The components are operational during these tests. No zero or span adjustments will be allowed during the tests. The acceptance criteria for instrument sensors are:

1) During the test the calibration must not change more than ±5%.

2) The post-test calibration must be within ±1-1/2% of the pre-test values.

The containment fan cooler motors are qualified in accordance with IEEE-334-1971, "IEEE Trial-Use Guide for Type Tests of Continuous-Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations."

Equipment, other than the containment fan cooler motors, is qualified for the accident radiation level by analysis. This analysis addresses the changes in properties of the component materials under radiation exposures equivalent to the specified conditions and establishes that these effects will not significantly affect performance characteristics. The fan cooler motors were tested as described in Appendix 3A.

b) Category I-C and I-D Equipment

This equipment is rated for continuous operation under the specified conditions. Since the extremes of the temperature, pressure and humidity conditions are within the design ratings of standard commercial components, qualification is accomplished by requiring vendor certification of the required capability and supporting documentation of test results or operating experience on similar equipment.

It is not technically feasible to predict the long term cumulative effects on equipment performance due to operation under normal service conditions which include low level radiation. Thus, neither type testing nor analysis is adequate to qualify the equipment for the long term environment.

The cable associated with category I-C and I-D equipment is purchased to the same specifications as category I-B equipment and meets the qualification criteria for I-B cable.

The requirement for meeting the specified performance requirements from the most degraded conditions resulting from long term environmental effects is met by limiting the amount of degradation allowed. The periodic test and calibration program described in Chapter 16 will verify the capability of the equipment to meet the original performance requirements. Periodic testing will allow detection of any gradual equipment deterioration and effectively requalifies the equipment for the shortterm operational period between tests.

3.11.3 QUALIFICATION TEST RESULTS

In cases where the information in this section overlaps with 10 CFR 50.49, the Equipment Qualification List (EQ) list and Documentation Packages (Doc Pacs) supersede the information provided here. Therefore, the EQ List and appropriate Doc Pacs should be consulted prior to the use of the information in this section.

	Results Contained in Appendix 3A
a) Category I-A	
Containment fan coolers	Westinghouse Electric Corporation. "Fan Cooler Motor Unit Test", WCAP 7829 (note that this is a proprietary report except for the abstract).
Containment fan cooler electrical cables	Cable vendors have performed tests on equivalent cable. See Table 3A-1.
Containment electrical penetration assemblies	Four reports were prepared by Gulf Electronics:
1) Measurement channel shield cable	E-115-173 "Instrumentation Prototype Penetrations", ELD 261-1004, Test Report.
2) 120 v power and 125 v or less dc control	E-115-176 "Low Voltage Power & Control Prototype Penetrations", ELD 261-1003, Test Report.
3) 120 v and 480 v power	E-115-180 "Low Voltage Power Prototype Penetrations", ELD 261-1002, Test Report.
4) 6.9 kv power	E-115-181 "High Voltage Power Prototype Penetrations", ELD-261-1001, Test Report.
Electrical Penetration	Type Tests by Raychem
Termination and Connectors	
Electrical Penetrations (C3, C6, C8, C9, D3 & D1O)	Certificate of Compliance received from vendor referencing prototype test results.
Electrical Penetration Replacement Modules	Reports prepared by Conax Corporation - IPS-353.10 Rev B - Qualification Report for a 42 conductor #18 AW Conax Low Voltage Service Classification Conductor Feed through Assembly - IPS-585.2 Rev B - Qualification of Instrumentation Services Classification Electrical Penetration - IPS-585.3 Rev D - Qualification Test Report for a Low Voltage Power and Control Services Classification Electric Penetration Assembly for Class 1E Service In BWR & PWR Containment structures

- IPS-590 Rev 0 - Test Summary IEEE 383 Vertical Tray Flame Test on Kapton Insulation
- IPS-1229 Rev 0 - Qualification Report for Feedthrough/Adapter Module Assemblies for Gulf General Atomic Penetrations
- IPS-325 Rev E - Qualification Material Test Report for Materials Used in Conax Nuclear Products for Service in Nuclear Power Generating Stations
- IPS-694 Rev 0 - Heat Transfer Analysis for Electric Penetrations
- IPS-1230 Rev 0 - Structural Integrity of GA Penetrations and the Conax Adapter Module

b) Category I-B

Pressurizer pressure sensors, steam generator pressure sensors, containment pressure sensors	Type test results are in a format consistent with the intent of IEEE-323-1971 although some of the specific details may vary since the testing was accomplished prior to issuance of the standard. Fisher & Porter Co. "Special High Temperature Steam Application", Report #DP-2004-51-B-006, 1968.
Letdown line isolation valves	No qualification test available.
Containment radiation sensors	A quality assurance procedure is being prepared by Victoreen. "Containment Monitor Test Procedure," PT-64.

Electrical cables associated	Cable vendors have performed tests on equivalent cable. Refer to Table 3A-1.

c) Category I-C and I-D

The equipment in this category (except for the electrical trip switchgear) are standard instrument packages for which the vendors have specifically certified the individual equipment's capability to meet the operating environments as delineated in the purchase specification. These conclusions are based on a combination of type testing and proven operating capability in similar environments in industrial and previous nuclear power plant applications.

Electrical trip switchgear	General Electric has performed tests on equivalent equipment. A test report has been submitted.

3.11.4 LOSS OF VENTILATION

3.11.4.1 Control Room Air Conditioning and Ventilation Systems

Controls and electrical equipment necessary for safe plant shutdown are located in the control room and are classified as category I-D equipment. The control room is normally air conditioned to provide operators and equipment optimum environmental conditions.

Assuming a loss of off-site power, control room air conditioning will be reinstated by automatically loading them on the emergency buses powered from the diesel generators. Ventilation is continuously available even after a loss of off-site power through redundant outside air intake systems. The outside air intakes are used as required when air conditioning is available and are always available during normal and accident operation. A CIS automatically isolates the intakes during a LOCA. Refer to Section 9.4.1 for an analysis of control room environmental conditions during a LOCA.

3.11.4.2 Emergency Switchgear Room Ventilation System

Redundant emergency switchgear for safety related equipment is located in separate rooms in the reactor auxiliary building. The switchgear rooms are provided with a ventilation system with redundant fans. A detailed description is given in Section 9.4.2.

The switchgear equipment is designed to function under conditions of 104 F and 90 percent relative humidity for an extended period.

The ventilation equipment for the emergency switchgear room is designed to seismic Class I requirements and each switchgear room fan is supplied with on-site emergency power from its associated standby diesel generator upon loss of off-site power.

3.11.4.3 Emergency Core Cooling System Pump Room Ventilation System

The ECCS pump room ventilation system is described in detail in Section 9.4.3.

Redundant ECCS pumps are located in separate compartments in the reactor auxiliary building. Under normal operating conditions, the ECCS pump compartments are maintained at or below 104°F by normal ventilation system operation. During such periods, the ECCS pumps are not required to operate. The ventilation system is designed to maintain pump room temperature below 104°F with the ECCS pumps and associated electrical equipment operating at full design capacity under LOCA conditions or pump testing periods. The ECCS equipment is designed to operate satisfactorily, in an environment of 120°F and 100 percent relative humidity for 6 hours and at or below 150°F and 90 percent relative humidity for an extended period.

The ECCS pump room ventilation system is designed to meet seismic Class I requirements and on-site emergency power will be supplied to each redundant fan from its associated standby diesel generator upon loss of off-site power. No single active failure can result in loss of ventilation to more than one ECCS pump compartment.

3.11.5 ELECTRICAL TERMINATIONS INSIDE CONTAINMENT

The staff in the attachments to its August 19, 1975 letter at item 5 expressed concern regarding the use of post-LOCA qualified and nonqualified terminations at the terminal boxes within containment. In the enclosure to its October 10, 1975 letter at item 1 the staff identified requirements for additional information concerning this matter following review of information submitted in Amendment 50.

The following sections present the additional information requested by the staff, describe the detailed evaluation of the existing design and present the basis and justification for proposed modifications which, in the applicant's view, satisfy the staff's concern regarding adverse interaction between safety related and non-safety related terminations inside containment.

Table 3.11-2 is a summary of Sections 3.11.5.1 through 3.11.5.3 and is maintained for historical reference.

3.11.5.1 Physical Arrangement and Separation Philosophy of Terminal Boxes

Electrical terminations inside containment are made in a set of terminal boxes located adjacent to the electrical penetration area. The arrangement is shown on Figure 8.3-12.

There are a total of 14 boxes in use (and one spare), 10 of which contain both safety related (Class 1E) and non-safety related cables. The four remaining boxes contain only non-safety related circuits. All safety related terminations are made with post-LOCA qualified splices or connectors. Non-safety related terminations are in general made on General Electric Type EB-5 or CR-2960 terminal blocks, although there are some such circuits which are terminated with qualified splices or connectors. The non-safety related spliced circuits are those which emanate from a penetration module which contains at least one safety related conductor. In such cases, all conductors from the penetration module are spliced, both safety related and non-safety related. Qualification of the terminal blocks is discussed in Section 3.11.5.3 below. There are a total of about 3800 terminations in use of which approximately 2700 are currently made on terminal blocks.

Cable terminations are grouped in separate boxes according to:

(1) General functional requirements

(2) Safety channel designation

(3) Circuit voltage level

The terminations in each box are separated from those in other boxes by the steel barriers comprising the box enclosure. Four of the boxes (I spare) are double boxes with a vertical steel barrier as part of the box enclosure. Double boxes contain only A system or B system circuits. Further physical separation is provided between safety system channels by an additional vertical steel fire barrier (See Figure 3.8-8AA) which separates those boxes containing A system control, power and monitoring circuits from B system control, power and monitoring circuits. Double physical separation is thus provided for all safety related terminations, including both the individual terminal box barriers and the penetration area separation barrier. The separation scheme is shown on Figure 8.3-12 and is described as follows:

<u>East Side of Penetration Barrier Wall (A System)</u>

1. Measurement Channel and Monitoring Instrumentation Circuits (low voltage)

   TB L105 (MA) - Measurement Channel MA, Safety System SA, Non-Safety related A

   TB L103 (MC) - Measurement Channel MC, Safety System SA, Non-Safety related A

   TB L104 (MA) - Non Safety Related A

2. Small Power and Control Circuits - (125 dc or 120 vac)

   TB C107 (SA) - Safety System SA, Non-Safety Related A

   TB C109 (SA) - Safety System SA, Non-Safety Related A

   TB C108 (SA) - Non Safety Related A

3. Power Supply Circuits - (120 - 480 vac)

   TB M101 (SA) - Safety system SA, Non-Safety Related A

   TB M102 (SA) - Non Safety Related A

<u>West Side of Penetration Barrier Wall (B System)</u>

1. Measurement Channel and Monitoring Instrumentation Circuits (low voltage)

   TB L125 (MB) - Measurement Channels MB, Safety System SB, Non-Safety Related B

   TB L123 (MD) - Measurement Channels MD, Safety System SB, Non-Safety Related B

2. Small Power and Control Circuits - (125 v dc or 120 vac)

   TB C127 (SB) - Safety system SB, Non-Safety Related B

   TB C129 (SB) - Safety system SB, Non-Safety Related B

   TB C128 (SB) - Non Safety Related B

3. Power Supply Circuits (120-480 vac)

   TB M123 (SB) - Safety System SB

As can be seen the circuits are grouped in separate boxes by safety system, function and voltage level. Low voltage measurement circuits are segregated from those circuits which carry significant voltage sources. Control circuits are further separated from large power circuits for equipment inside containment. This provision limits the extent of interaction between circuits by segregating circuits according to function and capacity.

No box contains circuits belonging to two redundant safety systems. Four channel separation is maintained among the measurement channel circuits (MA, MB, MC and MD) which provide input to protection system logic.

3.11.5.2 Functional and Failure Mode Evaluation of Electrical Terminations

To evaluate the potential for possible adverse interaction between the qualified spliced or connector terminations and the terminal block terminations a complete review of the safety functions of all circuits for normal operation, accident and post-accident conditions was conducted. Individual circuit failure modes and potential common mode failure mechanisms were evaluated to determine whether the physical separation and independence provided between safety circuits could be compromised or rendered ineffective as a result of failure of the terminal blocks upon which nonsafety related terminations were made.

The evaluation considered normal operation and those events which would occur in conjunction with normal environmental conditions in addition to those accidents which are accompanied by severe environment effects. Table 3.11-2 provides a summary listing of all safety related circuits and their safety functions under all postulated events. The following sections provide a discussion of the basis of the evaluation.

3.11.5.2.1 Normal Environmental Events and Conditions

For normal operation and those accidents or events that do not involve release of primary or secondary coolant to the containment, the electrical circuits inside containment are not exposed to unusual environmental conditions. The pressure, temperature, humidity and radiation levels do not differ significantly from that which will be present throughout the operating life of the plant and there are no severe transient changes in any of these environmental parameters. The events in this category include normal operation and those anticipated reactor transients discussed in Section 15.2 and 15.3 (e.g., control rod drop, loss of coolant flow) which require the safety function of electrical circuits to trip the reactor and operate safety related equipment to mitigate the effects of the accident and/or achieve subsequent plant shutdown and cooldown.

For this category of events, two hundred and twenty (220) conductors serve to provide input signals to reactor protection system (RPS) and engineered safety feature actuation system (ESFAS) logic. These conductors are separated into the four measurement channel groups (MA, MB, MC, MD) and are located in four separate terminal boxes, each containing 55 of the conductors. These conductors provide safety functions during normal operation by monitoring plant parameters which would indicate occurrence of accidents or events requiring automatic protective action.

One hundred and ninety two (192) of the 220 conductors are required to function for some or all of the reactor transients evaluated in Chapter 15. These conductors provide automatic reactor trip signals on high power level, low coolant flow, thermal margin/low pressure and low steam generator level or pressure for transient events lasting in the order of seconds to several minutes. Once they have provided the necessary trip input signal, the circuits have no further automatic protective function since the RPS logic will seal in the trip signal outside containment. One hundred and forty (140) of these conductors may also be required for post-accident monitoring following a transient event in order to allow the operator to monitor primary and secondary conditions and to commence an orderly shutdown and cooldown if required following the event. These are the protection system measurement channels for reactor coolant temperature, pressurizer pressure, and steam generator pressure and level. In addition, six (6) conductors associated with the low range pressurizer pressure sensors and six (6) conductors for pressurizer level sensors provide post-accident and shutdown monitoring safety functions. These monitoring functions may be required for up to several hours following the event until the primary system can be depressurized to the point where shutdown cooling can be initiated, after which shutdown cooling system instrumentation located outside containment can be used to monitor plant conditions.

Since the events considered in this category are not accompanied by abnormal environmental conditions, a potential environmental common mode mechanism for adverse interaction between safety related and non-safety related circuits in more than one terminal box is not present. Environmental qualification of the terminal blocks is therefore not a concern for this category of events. The potential for common mode failure of non-safety related terminations due to seismic events has been eliminated by qualification of the terminal blocks for Design Basis Earthquake loading (see Section 3.11.5.3).

Since four channel separation is provided among the measurement channels as discussed above, no single fault on a non-safety related termination in any one terminal box can affect more than one measurement channel. A sufficient number of measurement channels would remain operable to provide automatic reactor trip input signals or post-accident monitoring functions if required. It is therefore concluded that the design of the safety related measurement channel and monitoring instrumentation circuits is adequate for normal operation and those reactor transients which do not involve severe environmental effects.

Six (6) conductors provide signals from pressurized pressure transmitters to shutdown cooling system isolation valve interlocks. The integrity of these circuits is necessary to permit removal of the interlock signal to allow openings of the isolation valve for normal shutdown and, if required, for plant shutdown and cooldown following reactor transients. Faults on these circuits could prevent opening of the valves. However, opening of these valves would not be required until several hours after the event and blocking conditions on this circuit could be defeated at the control board, if necessary, to allow operation of these shutdown cooling isolation valves. Accordingly the design of these six qualified terminations is considered adequate.

Eighteen (18) conductors provide local push button control of the four containment fan cooler units. Faulting of these circuits could interrupt fan cooler operation. No single fault on a non-safety related terminal block could result in complete loss of containment cooling due to the separation of safety systems in the terminal boxes. Even if containment cooling were interrupted during normal operation or during or following any of the reactor transients in this category, the plant could be maintained in a safe condition. The design of these 18 qualified terminations is therefore considered to be adequate for these events.

Twenty-eight (28) conductors provide limit switch control for the shutdown cooling isolation valves. These valves are required for normal shutdown and for shutdown following reactor transients. As discussed in Section 3.11.5.2.2 there are certain highly unlikely faults which could prevent remote valve operation. However for the category of events which do not involve severe environment effects, personnel access to the containment is permissible and provides a means of manually operating the valves despite electrical circuit faults which would prevent remote operation. Therefore, the design of these qualified terminations is considered adequate for normal operation and reactor transients.

3.11.5.2.2 Severe Environmental Events and Conditions

Accidents involving release of primary reactor coolant or secondary (main steam or feedwater) system coolant are accompanied by significant increases in containment pressure, temperature and humidity. Loss of coolant accidents (LOCA) can also, depending on coolant radioactivity inventory and extent of fuel damage, result in large increases in containment radiation dose levels. The safety functions of all electrical circuits during and following LOCA and secondary line break accidents (SLBA) have been evaluated to identify the short term and long term functional requirements for these severe environmental events.

The various LOCA related junctions fall into several categories each of which is discussed here. Forty (40) conductors actuate SIAS, CIS or CSAS and are not required thereafter. Once SIAS, CIS or CSAS are actuated they lock in and any long term effect of the LOCA environment on the sensing instruments will not adversely affect the safety logic. Since the SIAS, CIS or CSAS are actuated in the order of seconds to a few minutes and conceivable environmental effects are long term, the design with regard to these 40 qualified terminations is considered acceptable.

Thirty-one (31) conductors provide indication of containment isolation valve position. Any failure mode of these indicating circuits will not adversely affect valve position. Thus any long term degradation postulated to occur to the qualified terminations results only in loss of indication. These valves are not required to re-open once closed. Since containment integrity is not threatened, the design with regard to these qualified 21 terminations is considered adequate. Eight (8) conductors provide containment air temperature indication for post-LOCA monitoring for the first 5 or 6 hours following LOCA initiation. Thereafter, the sump water and air are in thermodynamic equilibrium, thus either provides long term containment temperature indication. Since the short term temperature transient is relatively insensitive to large fluctuations in pressure, the containment temperature is well represented by the saturation temperature associated with the partial pressure of steam and the partial pressure of air can be determined with acceptable accuracy, containment temperature can be easily determined without direct measurement. Loss of containment air temperature indication results in no public health and safety considerations (see Section 7.5.2.2.2). Accordingly the design of these 8 qualified terminations is considered adequate.

Eighteen (18) conductors provide local push button control of the four fan cooler units. A failure mode effect criticality analysis of this circuit indicates that a postulated short could cause the motor to stop. Occurrence of such a short at the qualified terminations, although conceivable, is not likely. However, if such a short is postulated to occur, the public health and safety is not adversely affected. The reason is that three 100 percent containment cooling systems have been provided. The four fan cooler units comprise one 100 percent system. Two 100 percent spray systems are also available and one containment spray subsystem in conjunction with two containment fan coolers comprise a 100 percent system. Thus if a short in each fan cooler pushbutton circuit is postulated to occur coincident with a single failure a single

100 percent spray system is still available to provide the containment cooling function. It must be noted that the fan cooler motor stop can be defeated by disrupting the local pushbutton circuit at the MCC, thereby allowing the fan cooler units to be restored to service. Accordingly, the design of these 18 qualified terminations is considered adequate.

Local push button station cables for valves V3480, V3481, V3651 and V3652 have been disconnected per PC/M 84133.

Twelve (12) conductors provide power to pilot operated, air to open, isolation valves. Interruption of power to the solenoid pilot valve vents the air causing the valves to close. An analysis of the solenoid pilot valve circuitry indicates that no failure within the circuit will cause the pilot valve to direct air to open the isolation valve. In addition an analysis of the air system indicates that the 24 SCFM air receiver will lose its contained energy in about one hour - motive air force to reopen a valve will be lost in less than one hour. Thus, if postulated effects external to the solenoid pilot circuitry are assumed, the motive power to reopen the isolation valve is not available. Accordingly, the design of these 10 qualified terminations is considered acceptable.

Twenty-eight (28) conductors provide limit switch control of the shutdown cooling system isolation valve. A failure mode effect criticality analysis indicates that all failure mechanisms that could be reasonably postulated will not adversely affect valve operability. However, an extremely remote event affecting 4 of the 28 conductors can be conceived that would adversely affect valve operability, namely, failure of the qualified splice with the simultaneous application of an external power source. The small likelihood of occurrence of this event, the qualification discussion infra and the redundancy provided (two separate lines, each with two valves) lead to the conclusion that the design of these 28 qualified terminations is acceptable.

Fifty-one (51) conductors supply power to;

a) H_2 analyzer isolation valves (solenoid operated)

b) Shutdown cooling isolation valves (motor operated)

c) Pressurizer auxiliary spray valves (solenoid operated)

Of these valves (a) are not required for the protection of the public health and safety, but their availability is useful, but not mandatory, for following the course of the accident. Valves (b) and (c) are included because their use long term may be required to accommodate the Staff's concern regarding boron precipitation. Further discussion is provided below to demonstrate that LOCA environment will not adversely affect the operability of these valves, as well as, the 4 limit switch conductors mentioned supra. There are 17 cables and 55 conductors out of the 3800 conductors that are relevant to the Staff's expressed concern. These cables are terminated in the control and power terminal boxes.

In addition to the safety functions described above for LOCA conditions, there are certain conductors which are required to perform short or long term safety functions for SLBA events.

Twenty-four (24) conductors actuate MSIS on low steam generator pressure. These same 24 conductors as well as one hundred and twenty-eight (128) additional conductors provide post-SLBA functions in monitoring primary and secondary conditions to permit cooldown and depressurization to the point where shutdown cooling can be initiated after which shutdown cooling system instrumentation located outside containment can be used to monitor plant conditions. The post accident monitoring function is required only for a short term period of up to several hours. The conductors involved are located in terminal boxes containing only low voltage instrumentation circuits which are not a potential source of large current faulting which could result in adverse interaction between safety related and non-safety related terminations. In addition, the terminal blocks involved have been shown to be qualified for at least up to four days in a temperature , and humidity environment more severe than post-SLBA functions. It may be concluded therefore that the potential for adverse interaction within the measurement channel boxes for the period of required post-SLBA safety function (several hours) is negligible for these 152 conductors.

The circuits associated with pressurizer and containment measurement channels, fan cooler units and shutdown cooling isolation valves are also required to perform similar safety functions for SLBA events as for LOCA events. The discussion given above for these circuits under post LOCA conditions is equally applicable to SLBA conditions.

3.11.5.3 Qualification of Terminal Blocks

The terminal cabinets contain GE EB-5 terminal blocks and are located in the containment to serve as connection and array boxes for the containment penetration conductor pigtails (see Figure 8.3-12).

Typically, cables which have a safety related function during normal, incident or accident conditions are spliced with Raychem splices or connected with qualified connectors and routed through the boxes to their distributive tray system whereas, non-safety cables (identical to safety cables) pass through the boxes and are terminated on GE EB-5 terminal boards for connection, array and continuation to the required tray distribution systems.

The terminal blocks used are General Electric Type EB-5 and CR-2960 rated 600 volts, 30 amperes, and 600 volts, 150 amperes respectively. These blocks are specifically designed for applications of terminations of many wires in a permanent, orderly fashion.

The boards are prefabricated, equipped with washer head binding screws for circuit wire connections and marker strip for circuit identification. Terminal cabinet mounting is provided with bolts. The block will accommodate wire sizes No. 18 to No. 10 and No. 12 to No. 2 inclusive and are arrayed within the cabinets as required in groups of 2 to 18 points.

The terminal boards and cabinet are in conformance with applicable industry standards as follows:

ANSI C19.3 Industrial Control Apparatus - General

ANSI C19.4 Industrial Control Apparatus - Enclosures

The standards define the electrical and mechanical criteria for arrays of terminals within enclosures. Criteria addressed by these standards include voltage, current, clearances, creepage, service conditions, thermal ratings, short circuit ratings etc. and testing parameters. The GE EB-5 terminal board and associated boxes in which they are mounted satisfy the intent of the application and associated standards.

Specific design criteria for this application are met as follows:

a) General Design Criteria 1 (Quality Standards and Records)

The terminal blocks and cabinets were purchased and erected in accordance with industry standards, Ebasco Engineering Guides and installed with Quality Compliance, Quality Assurance and Construction Procedures applicable to junction and terminal boxes.

b) General Design Criteria 2 (Design Basis For Protection Against Natural Phenomena)

1) Seismic Qualification

(a) By analysis directly verifying the following;

(1) The structural integrity (including support mechanisms) for both the Reactor Auxiliary Building and containment penetration wireways/terminal boxes will not be violated during design basis earthquake (DBE).

(2) Penetration wireways/terminal boxes design is such that the natural frequency of the loaded boxes are not in resonance with the postulated seismic forcing function.

(3) Analysis verifies that the boxes are adequately designed such that seismic accelerations are withstood.

(4) Acceleration response spectra utilized in the analysis represents the floor response spectra for the appropriate floor elevation.

(5) All boxes are fabricated by a vendor who has been qualified to manufacture material for seismic Class I service.

(b) By similar comparison, the General Electric EB-5 and CR-2960 terminal boards used on St. Lucie 1 were seismically tested as part of the Diesel Generator Control Panel at higher seismic levels than that which would be experienced inside the containment. Additionally, these terminal boards are presently in use in other operating nuclear units for Class 1E service.

2) Hurricanes, tornadoes, floods, etc - not applicable as these boxes are applied within the containment annulus area.

c) General Design Criteria 3 (Fire Protection)

Redundant and independent terminal boxes are used for each safety system termination as shown in Figure 8.3-12. Therefore, any fire in one will not have an effect on the redundant system. Redundant safety systems are separated by steel barriers. (See discussion on physical separation in Section 8.3.1.2.3.)

Further, the cabinets are made of steel which is non-flammable and the blocks will not support combustion upon extinguishment of external flame source (see discussion below concerning block environmental testing).

d) General Design Criteria 4 (Environmental and Missile Design Basis)

1) Temperature and Humidity

The following is a description of a test, which is being conducted by FPL at their Materials Test Laboratory.

Test Objective

The objective of the test is to demonstrate that GE type EB-5 and CR-2960 type terminal blocks can function in a high temperature steam atmosphere similar to that which might be experienced during Post-LOCA conditions at the St. Lucie Plant.

Test Set Up

Three blocks of each type were mounted in a box with spacing similar to that at St. Lucie Plant. Space heaters were installed in the box to assure that the maximum environmental temperature would be approximately 250°F.

Test voltages of 120 volts AC, 125 volts DC, and 480 volts AC were applied to the terminals of the EB-5 terminal blocks. Three-phase, 480 volt AC was applied to the CR-2960 blocks. The CR-2960 is used only for 480 volt three phase service in the containment penetration terminal boxes.

A boric acid solution of 2000 ppm buffered to ph of 8.5 with NaOH was sprayed on the blocks for 15 minutes. Steam made from the above boric acid solution was then introduced into the box. At the end of four days of steam and temperatures of approximately 250°F, there were no signs of electrical failures (shorts or opens of the terminations) nor were there any signs of physical damage. No arcing or other phenomena which might cause fire or otherwise affect surrounding cabling, or otherwise inadvertently affect the surrounding blocks were observed.

In addition to the foregoing environmental testing the blocks were tested for their fire resistant properties.

A potential of 12000 volts was applied to a point on the block and an arc was drawn but the block did not ignite.

A burnzomatic torch flame was applied to another block. The block would burn while the flame was applied but immediately went out when the torch was removed.

The foregoing testing indicates that the terminal blocks used in the penetration termination cabinets would not adversely affect the qualified terminations or surrounding cables during Post-LOCA conditions.

2) Radiation

A Buchanan Terminal Block (Catalog Number 2B112N) which is similar, dimensionally to a General Electric EB-5 and a GE CR141 terminal block, were exposed to a $1.1x10^7$ Rads/hr source for 13.6 hours (accumulated dose $1.5x10^8$ Rads) by a penetration vendor.

Due to the high exposure rate the Buchanan Terminal Block suffered minor malformation from excess temperature in the range of 400°F. The General Electric terminal block had no malformation.

Upon removal of the terminal blocks from the test chamber Hi Pot Tests were were run with both terminal blocks performing satisfactorily.

The GE CR151 terminal block is materially similar to the GE EB-5 and CR2960 blocks.

The General Electric Terminal Block CR151 has successfully passed a LOCA test as part of this penetration qualification program which included:

Time	Temperatures(°F)	Humidity (%)	Pressure (Psig)
0 to 45 Sec	340	100	62
45 sec to 6 hours	340	100	35
6 hours to 12 hours	320	100	35
12 hours to 24 hours	200	100	25
24 hrs to 100 days	200	100	25

The accumulated radiation dose was 1×10^8 Rads which satisfies the Category I-A radiation requirements for St. Lucie.

Further, it should be noted GE EB-5 and CR2960 terminal boards are used within existing operating units, inside and outside the containment.

e) General Design Criteria 17 - (Electrical Power Systems)

Sufficient independence, redundancy and testability is provided within the design as evidenced on the terminal box location and conductor array drawings.

Each safety system uses separate metal enclosed boxes, is separated and is accessible (removable covers) for circuit testing/checkout.

On the basis of the foregoing discussion it is concluded that no undue risk results from the in situ terminal board installation. Tests indicate that the LOCA or SLBA conditions will not cause an unacceptable failure mechanism during the initial phase of long term post LOCA operation.

3.11.5.4 Proposed Design Modification

The functional and failure mode evaluation discussed in Section 3.11.5.2 above identified certain circuits required for normal environmental and severe environmental events and conditions. The physical separation provided among safety related circuits within the terminal boxes provides assurance that any single event within a terminal box cannot result in loss of capability to respond to and mitigate the consequences of accidents and achieve safe shutdown. Seismic qualification of all terminations, both those spliced and those made on terminal blocks, eliminates seismic events *as* a potential mechanism for common mode failure of non-safety related circuits in more than one terminal box. The concern for adverse interaction between safety related and non-safety related terminations can therefore be limited to SLBA or LOCA events involving severe environmental conditions.

Circuits required to perform short term safety functions for these events are low voltage measurement channel and monitoring instrumentation. The safety functions of these circuits are limited to RPS and ESFAS actuation (a matter of seconds or minutes) and post-accident monitoring (up to several hours). As discussed in Section 3.11.5.3, the qualification data available for the GE EB-5 and CR-2960 blocks provide reasonable assurance that the terminal block will maintain their functional integrity. In any event, the low voltage circuits are segregated in separate terminal boxes from the high energy control and power circuits. The low voltage circuits have insufficient energy associated with them to be of concern. This is quantified by considering short circuit current load. Table 3.11-3 summarizes the failure potential associated with those low voltage circuits, which are terminated on terminal blocks. The circuits involved, with one exception, carry low voltages and are connected to high resistance or current limiting external circuits outside containment. The exception being the annunciation circuits which, although operating it 125 v dc, have a high resistance external circuitry which limits short circuit current to 0.005 amp. In all cases the maximum short circuit currents are less than 1 amp and well below the minimum 5 amp current carrying capacity of the associated wiring. For these reasons, it is concluded that no design modifications are required within those boxes containing low voltage measurement channel and instrumentation circuits. In addition, no protection is required for annunciator circuits.

A number of circuits were identified in Section 3.11.5.2.2 which are required to perform long term safety functions for severe environmental events. These circuits are terminated in both control and power circuit boxes which contain voltage sources from 120v to 480v. It may be postulated that in the long term period following a SLBA or LOCA event a potential for adverse interaction between the required circuits and energized non-safety circuits is possible. Tile potential exists due to the energy levels associated with these circuits. To satisfy the stall's concern on this matter, all terminations located in control and power circuit boxes (with the exception of annunciator circuits) which could be energized followings a SLBA or LOCA event have been provided with splices or connectors which are fully qualified for the post-LOCA environmental conditions. Implementation of this modification involved replacing approximately 300 existing connections which had been made on terminal blocks with qualified splices or connectors. The terminal boxes involved include:

Small Power and Control Circuit Boxes

TB C107 (SA)
TB C109 (SA)
TB C108 (SA)
TB C127 (SB)
TB C129 (SB)
TB C128 (SB)

Power Circuit Boxes

TB M101 (SA)
TB M102 (SA)
TB M123 (SB)

All terminal block connections have been eliminated from the power and control boxes with the exception of the annunciation circuits and non-safety related CEDM Holding coil circuits which are de-energized at the reactor trip switchgear outside containment immediately upon reactor trip and thus provide no source of interaction during the subsequent period following trip). These CEDM circuits and the non-safety related circuits of the type identified in Table 3.11-3 are the only remaining terminations made on terminal blocks.

This design modification effectively eliminated the potential for adverse interaction between safety-related and non-safety related terminations by eliminating the high energy circuits from the harsh LOCA/SLBA environment.

REFERENCES FOR SECTIONS 3.11

1. R. E. Uhrig (FPL) to D. G. Eisenhut (NRC) Re: St. Lucie Unit 1, Docket No.50-335, NUREG-0578 Short Term Requirements, L-80-17 dated 1/11/80.

2. R. E. Uhrig (FPL) to J. P. O'Reilly (NRC) Re: RII: JPO Docket No. 50-335, IE Bulletin 79-01B, L-80-167, dated 6.2.80.

3. R. E. Uhrig (FPL) to J. P. O'Reilly (NRC) Re: RII: JPO Docket No. 50-335, IE Bulletin 79-OIB, L-80-259 dated 8/6/80.

4. R. E. Uhrig (FPL) to J. P. O'Reilly (NRC) Re: Turkey Point Units 3 & 4, St. Lucie Unit 1, Docket Nos. 50-250, 50-251 and 50-335, Environmental Qualification of Electrical Equipment, L-80-363 dated 10/31/80.

5. R. E. Uhrig (FPL) to J. P. O'Reilly (NRC) Re: RII: JPO Docket No. 50- 335, IE Bulletin 79-01B (Supplement 3), L-81-42 dated 2/6/81.

6. R. E. Uhrig (FPL) to R. A. Clark (NRC) Re: St. Lucie Unit 1, Docket No. 50-335, Environmental Qualification of Safety Related Electrical Equipment, L-81-442 dated 10/8/81.

7. R. E. Uhrig (FPL) to D. G. Eisenhut (NRC) Re: St. Lucie Unit 1, Docket No. 50-335, Radiation Dose Map Development (Rev. 1-6/24/83), L-83-409 dated 7/15/83.

TABLE 3.11-1

SPECIFIED ENVIRONMENTAL CONDITIONS
Historical

Environmental Design Category	Time	Temperature	Pressure	Humidity	Radiation
I-A	0 - 2 hr	270 F	44 psig	100%	2×10^6 R/hr
	2 - 24 hr	240 F	27 psig	100%	1×10^6 R/hr
	24 - 72 hr	190 F	10 psig	100%	3×10^3 R/hr
	72 hr - 31 day	150 F	5 psig	100%	3×10^3 R/hr
	31 day - 180 day	130 F	1 psig	100%	10 R/hr
I-B	15 min	270 F	44 psig	100%	7.6×10^5 R
I-C	Continuous	120 F	Atmos	40%	1 R/hr
I-D	Continuous	120 F	Atmos	40%	Negligible
	1 hr	135 F	Atmos	90%	Negligible

TABLE 3.11-1A

ELEVATION OF MOTOR OPERATED VALVES

INSIDE CONTAINMENT

Valve Number	Figure	Elevation
V1403	5.1-3	>80'
V1405	5.1-3	>80'
V3480	6.3-2	32'10"
V3481	6.3-2	35'6"
V3614	6.3-2	23'10"
V3624	6.3-2	23'10"
V3634	6.3-2	24'10"
V3644	6.3-2	23'2"
V3651	6.3-2	29'0"
V3652	6.3-2	32'10"

TABLE 3.11-2

CIRCUITS REQUIRED FOR NORMAL ACCIDENT AND POST ACCIDENT

SAFETY RELATED FUNCTIONS LOCATED IN TERMINAL BOXES

circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment: Normal Operation	Normal Environment: Anticipated Transients Short Term	Normal Environment: Anticipated Transients Long Term	Severe Environment: SLBA/LOCA Short Term	Severe Environment: SLBA/LOCA Long Term
Terminal Box L105 (MA							
Neutron Flux Measurement Channel	Neut Det #1	1 (7)	P	RT	-	-	-
RCS Hot Leg 1A Temp Measurement Channel	TE 1112HA	1 (5)	P	RT,PAM	-	PAM*	-
RCS Hot Leg 1B Temp Measurement Channel	TE 1122HA	1 (5)	P	RT,PAM	-	PAM*	-
RCS Cold Leg 1A2 Temp Measure-ment Channel	TE 1112CA	1 (5)	P	RT,PAM	-	PAM*	-
RCS Cold Leg IBI Temp Measure-ment Channel	TE 1122CA	1 (5)	P	RT,PAM	-	PAM*	-
SG 1A Diff. Pressure Measurement Channel	PDT 1111 A	1 (3)	P	RT	-	-	-
SC 1B Diff. Pressure Measurement Channel	PDT 1121 A	1 (3)	P	RT	-	-	-
SG 1A Level Measurement Channel	LT 9013 A	1 (3)	P	RT,PAM	-	PAM*	-
SG 1B Level Measurement Channel	LT 9023 A	1 (3)	P	RT,PAM	-	PAM*	-
SG 1A Pressure Measurement Channel	PT 8013 A	1 (3)	P	RT,PAM	-	RT, MSIS,PAM*	-
SG 1B Pressure Measurement Channel	PT 8023 A	1 (3)	P	RT,PAM	-	RT, MSIS,PAM*	-
Pressurizer Pressure Measure-ment Channel	PT 1102 A	1 (3)	P	RT,PAM	-	SIAS,PAM*	-
Containment Pressure Measure-ment Channel	PT-07-2 A	1 (3)	P	-	-	RT,SIAS,CIS,CSAS	-
Containment Radiation Measure-ment Channel	GM-3	1 (4)	p	-	-	CIS	-

P = Protection System Monitoring Input
RT = Reactor Trip
PAM = Post Accident Monitoring
SIAS = Safety Injection Actuation Signal
MSIS = Main Steam Isolation Signal
CIS = Containment Isolation Signal
CSAS = Containment Spray Actuation Signal
SLBA = Secondary Line Break Accident
LOCA = Loss of Coolant Accident

* SLBA only

NOTE: This table was prepared specifically in response to safety review item 1 of the NRC letter from Parr to Uhrig dated 10/10/75. It has not been updated since.

See responses to IE Bulletin 79-01B References 2 through 6.

TABLE 3. 11- 2 (Cont 1 d)

Circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment: Normal Operation	Normal Environment: Anticipated Transients Short Term	Normal Environment: Anticipated Transients Long Term	Severe Environment SLBA/LOCA Short Term	Severe Environment SLBA/LOCA Long Term
Terminal Box L105 (MA)							
Containment Temp. Monitoring	TE-07-2A	1 (4)	-	-	-	PAM	-
Shutdown Cooling Isolation Valve Pressurizer Pressure Permissive Interlock	PT-1203	1 (3)	-	PAM	Maintain circuit integrity to prevent faults from blocking valve operation if required for plant shutdown. However, faults can be defeated at control board.	PAM*	Maintain circuit integrity to prevent faults from blocking valve operation if required for boron precipitation control. However, faults can be defeated at control board.

TABLE 3.11-2 (Cont'd)

Circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment: Normal Operation	Normal Environment: Anticipated Transients Short Term	Normal Environment: Anticipated Transients Long Term	Severe Environment SLBA/LOCA: Short Term	Severe Environment SLBA/LOCA: Long Term
Terminal Box L125 (MB)							
Neutron Flux Measurement Channel	Neut Det #2	1 (7)	P	RT	-	-	-
RCS Hot Leg 1A Temp Measurement Channel	TE 1112 HB	1 (5)	P	RT,PAM	-	PAM*	-
RCS Hot Leg 1B Temp Measurement Channel	TE 1122 HB	1 (5)	P	RT,PAM	-	PAM*	-
RCS Cold Leg IA2 Temp Measurement Channel	TE 1112 CB	1 (5)	P	RT,PAM	-	PAM*	-
RCS Cold Leg 1B1 Temp Measurement Channel	TE 1122 CB	1 (5)	P	RT,PAM	-	PAM*	-
SG 1A Diff. Pressure Measurement Channel	PDT 1111 B	1 (3)	P	RT	-	-	-
SC 1B Diff. Pressure Measurement Channel	PDT 1121 B	1 (3)	P	RT	-	-	-
SC 1A Level Measurement Channel	LT 9013 B	1 (3)	P	RT,PAM	-	PAM*	-
SG 1B Level Measurement Channel	LT 9023 B	1 (3)	P	RT,PAM	-	PAM*	-
SG 1A Pressure Measurement Channel	PT 8013 B	1 (3)	P	RT,PAM	-	RT,MSIS,PAM*	-
SC 1B Pressure Measurement Channel	PT 8023 B	1 (3)	P	RT,PAM	-	RT,MSIS,PAM*	-
Pressurizer Pressure Measurement Channel	PT 1102 B	1 (3)	P	RT,PAM	-	SIAS,PAM*	-
Containment Pressure Measurement Channel	PT-07-2B	1 (3)	P	-	-	RT,SIAS,CIS,CSAS	-
Containment Radiation Measurement Channel	GM-4	1 (4)	P	-	-	CIS	-
Containment Temp.Monitoring	TE-07-2B	1 (4)	-	-	-	PAM	-

*SLBA only

TABLE 3.11-2 (Cont'd)

circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment: Normal Operation	Normal Environment: Anticipated Transients Short Term	Normal Environment: Anticipated Transients Long Term	Severe Environment: SLBA/LOCA Short Term	Severe Environment: SLBA/LOCA Long Term
<u>Terminal Box L103 (MC)</u>							
Neutron Flux Measurement Channel	Neut Det #3	1 (7)	P	RT	-	-	-
RCS Hot Leg 1A Temp Measurement Channel	TE 1112HC	1 (5)	P	RT,PAM	-	PAM*	-
RCS Hot Leg 1B Temp Measurement Channel	TE 1122HC	1 (5)	P	RT,PAM	-	PAM*	-
RCS Cold Leg IA2 Temp Measurement Channel	TE 1112CC	1 (5)	P	RT,PAM	-	PAM*	-
RCS Cold Leg IBI Temp Measurement Channel	TE 1122CC	1 (5)	P	RT,PAM	-	PAM*	-
SG 1A Diff. Pressure Measurement Channel	PDT 1111 C	1 (3)	P	RT	-	-	-
SG 1B Diff. Pressure Measurement Channel	PDT 1121 C	1 (3)	P	RT	-	-	-
SG 1A Level Measurement Channel	LT 9013 C	1 (3)	P	RT,PAM		PAM*	
SG 1B Level Measurement Channel	LT 9023 C	1 (3)	P	RT,PAM		PAM*	
SC 1A Pressure Measurement Channel	PT 8013 C	1 (3)	P	RT,PAM		RT,MSIS,PAM*	-
SG 1B Pressure Measurement Channel	PT 8023 C	1 (3)	P	RT,PAM		RT,MSIS,PAM*	-
Pressurizer Pressure Measurement Channel	PT 1102 C	1 (3)	P	RT,PAM		SIAS,PAM*	-
Containment Pressure Measurement Channel	PT-07-2C	1 (3)	P	-	-	RT,SIAS,CIS,CSAS	-
Containment Radiation Measurement Channel	GM-5	1 (4)	P	-	-	CIS	-
Pressurizer Level Monitoring	LT-111OX	1 (3)	-	PAM	-	PAM*	-

*SLBA only

TABLE 3.11-2 (Cont'd)

circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment: Normal Operation	Normal Environment: Anticipated Transients Short Term	Anticipated Transients Long Term	Severe Environment SLBA/LOCA Short Term	SLBA/LOCA Long Term
Terminal Box L123 (MD)							
Neutron Flux Measurement Channel	Neut Det #4	1 (7)	P	RT	-	-	-
RCS Hot Leg 1A Temp Measurement Channel	TE 1112HD	1 (5)	P	RT,PAM	-	PAM*	-
RCS Hot Leg 1B Temp Measurement Channel	TE 1122HD	1 (5)	P	RT, PAM	-	PAM*	-
RCS Cold Leg lA2 Temp Measure-ment Channel	TE 1112CD	1 (5)	P	RT,PAM	-	PAM*	-
RCS Cold Leg 1B1 Temp Measure-ment Channel	TE 1122CD	1 (5)	P	RT,PAM	-	PAM*	-
SC 1A Diff. Pressure Measurement Channel	PDT 1111 D	1 (3)	P	RT	-	-	-
SC 1B Diff. Pressure Measurement Channel	PDT 1121 D	1 (3)	P	RT	-	-	-
SG 1A Level Measurement Channel	LT 9013 D	1 (3)	P	RT, PAM	-	PAM*	-
SG 1B Level Measurement Channel	LT 9023 D	1 (3)	P	RT, PAM	-	PAM*	-
SG 1A Pressure Measurement Channel	PT 8013 D	1 (3)	P	RT, PAM	-	RT, MSIS, PAM*	-
SG 1B Pressure Measurement Channel	PT 8023 D	1 (3)	P	RT, PAM	-	RT, MSIS, PAM*	-
Pressurizer Pressure Measure-Channel	PT 1102 D	1 (3)	P	RT, PAM	-	SIAS, PAM*	-
Containment Pressure Measure-ment Channel	PT-07-2D	1 (3)	P	-	-	RT, SIAS, CIS, CSAS	-
Containment Radiation Measure-ment Channel	GM-6	1 (4)	P	-	-	CIS	-
Pressurizer Level Monitoring	LT 1110Y	1 (3)	-	PAM	-	PAM*	-

*SLBA only

TABLE 3.11-2 (Cont'd)

circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment: Normal Operation	Normal Environment: Anticipated Transients Short Term	Normal Environment: Anticipated Transients Long Term	Severe Environment SLBA/LOCA Short Term	Severe Environment SLBA/LOCA Long Term
<u>Terminal Box L123 (MD)</u>							
Shutdown Cooling Isolation Valve Pressurizer Pressure Permissive Interlock	PT-1104	1 (3)	-	PAM	Maintain circuit integrity to prevent faults from blocking valve operation if required for plant shutdown. However, faults can be defeated at control board.	PAM*	Maintain circuit integrity to prevent faults from blocking valve operation if required for boron precipitation control. However, faults can be defeated at control board.

TABLE 3.11-2 (Cont'd)

Circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment			Severe Environment	
			Normal	Anticipated Transients		SLBA/LOCA	
			Operation	Short Term	Long Term	Short Term	Long Term
Terminal Box C107 (SA)							
Containment Isolation Valves (Air Operated) Pilot Solenoid Control	V2516 FCV-25-3 V3661	1 (2) 1 (2) 1 (2)	-	-	-	De-energize pilot solenoid to vent air from valve operator allowing valve to close	-
Containment Isolation Valves (Air Operated) Pilot Solenoid Control	V2516 FCV-25-3 V3661	1 (5) 1 (5) 1 (5)	-	-	-	Verify valve closure by position Indication	-
Containment Fan Cooler Isolation Button Control	HVS-1A HVS-1B	1 (4) 1 (4)	-	-	-	Maintain circuit integrity to prevent faults from interrupting fan cooler operation.	
Shutdown Cooling * System Isolation Valve Local Push Button Control	V3481 V3651	1 (5) 1 (5)			Maintain circuit integrity to pre-vent faults from blocking valve operation if re-quired for plant shutdown.	Not required Push button locked out at control room.	Maintain circuit integrity to pre-vent faults from blocking valve operation if re-quired for boron precipitation control.
Shutdown Cooling System Isolation Valve Limit Switch	V3481 V3651	1 (7) 1 (7)	Maintain circuit integrity to prevent faults from causing spurious valve opening	Maintain circuit integrity to prevent faults from causing spurious valve opening	Maintain circuit integrity to prevent faults from preventing valve opening if required for plant shutdown.	Maintain circuit integrity to prevent faults from causing spurious valve opening.	Maintain circuit integrity to prevent faults from preventing valve opening if required for boron precipitation control.
Pressurizer Auxiliary Spray Valve Solenoid Control	ISE-02-3	1 (2)	-	-	Operate valve if required for plant shutdown.	Close valve for containment iso-lation if open	Operate valve if required for boron precipitation control

TABLE 3.11-2 (Cont'd)

Circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment: Normal Operation	Normal Environment: Anticipated Transients Short Term	Normal Environment: Anticipated Transients Long Term	Severe Environment SLBA/LOCA: Short Term	Severe Environment SLBA/LOCA: Long Term
Terminal Box C129 (SB)							
Containment Isolation Valves (Air Operated) Pilot Solenoid Control	V 2515 FCV-26-1	1 (2) 1 (2)	-	-	-	De-energize pilot solenoid to vent air from valve operator allowing valve to close	-
Containment Isolation Valves (Air Operated) Position Limit Switch	V 2515 FCV-26-1	1 (5) 1 (3)	-	-	-	Verify valve closure by posi-tion indication	-
Containment Fan Cooler Local Push Button Control	HVS-ID	1 (5)	-	-	-	Maintain circuit integrity to prevent faults from interrupting fan cooler operation.	
Shutdown Cooling * System Isolation Valve Local Push Button Control	V3480 V3652	1 (5) 1 (5)	-	-	Maintain circuit integrity to pre-vent faults from blocking valve operation if re-quired for plant shutdown.	Not required. Push button locked out at control room	Maintain circuit integrity to pre-vent faults from blocking valve operation if re-quired for boron precipitation control.
Shutdown Cooling System Isolation Valve Limit Switch	V3480 V3652	1 (7) 1 (7)	Maintain circuit integrity to pre-vent faults from causing spurious valve opening	Maintain circuit integrity to pre-vent faults from causing spurious valve opening	Maintain circuit integrity to pre-vent faults from preventing valve opening If required for plant shutdown	Maintain circuit integrity to pre-vent faults from causing spurious valve opening	Maintain circuit integrity to pre-vent faults from preventing valve opening if required for boron precipi-tation control

* Note: Cables for pushbutton lifted per PCM 84133

TABLE 3.11-2 (Cont'd)

Circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment			Severe Environment	
			Normal Operation	Anticipated Transients		SLBA/LOCA	
				Short Term	Long Term	Short Term	Long Term
Terminal Box C127 (SB)							
Containment Isolation Valves (Air Operated) Pilot Solenoid Control	FCV-25-4	1 (2)	-	-	-	De-energize pilot solenoid to vent air from valve operator allowing valve to close	
Containment Isolation Valves Position Limit Switch	FCV-25-4 ISE-01-1	1 (5) 1 (3)	-	-	-	Verify valve closure by position indication	-
Containment Fan Cooler Local Push Button Control	HVS-1C	1 (5)	-	-	-	Maintain circuit integrity to prevent faults from interrupting fan cooler operation.	
Pressure Auxiliary Spray Valve Solenoid Control	ISE-02-4	1 (2)	-	-	-	Close valve for containment isolation if open	Operate valve if required for boron precipitation control
Containment Hydrogen Analyzer Isolation Valve Solenoid Control	SV-27-5 SV-27-6 SV-27-7	1 (5) 1 (5) 1 (5)	-	-	-	Not required. Valves normally closed	Operate valves for containment hydrogen sampling
RCP Bleed-off Isolation Valve Solenoid Control	ISE-01-1	1 (2)	-	-	-	De-energize to close valve	Maintain circuit integrity to prevent valve opening
Terminal Box C109 (SA)							
Containment Hydrogen Analyzer Isolation Valve Solenoid Control	SV-27-1 SV-27-2 SV-27-3 SV-27-4	1 (5) 1 (5) 1 (5) 1 (5)	-	-	-	Not required. Valves normally closed	Operate valves for containment hydrogen sampling
Terminal Box M101 SA							
Shutdown Cooling System Isolation Valve Power Supply	V3481 V3651	1 (3) 1 (3)	-	-	Provide power for valve operation if required for plant shutdown.	Not required. Valve normally closed. Power locked out at control room.	Provide power for valve operation if required for boron precipitation control.

TABLE 3.11-2 (Cont'd)

Circuit Function	Primary Device	No. Cables (No. Cond.)	Normal Environment: Normal Operation	Normal Environment: Anticipated Transients: Short Term	Normal Environment: Anticipated Transients: Long Term	Severe Environment: SLBA/LOCA: Short Term	Severe Environment: SLBA/LOCA: Long Term
<u>Terminal Box M123 (SB)</u>							
Shutdown Cooling System Isolation Valve Power Supply	V3480 V3652	1 (3) 1 (3)	-	-	Provide power for valve operation if required for plant shutdown	Not required. Valve normally closed. Power locked out at control room	Provide power for valve operation if required for boron precipitation control

TABLE 3.11-3

LOW VOLTAGE TERMINATION FAILURE POTENTIAL

Device	Cable Size & Current Carrying Capacity	Device Operating Voltage	Circuit Primary Protective Fuse	Max. Current In Shorted Cable	Failure Effects on Adjacent Terminations
Level Transmitter	#16 AWG 5A MIN	45V DC Max	1/8 A	0.2A	None
Pressure Transmitter	#16 AWG 5A MIN	45V DC Max	1/8 A	0.2A	None
Flow Transmitter	#16 AWG 5A MIN	45V DC Max	1/8 A	0.2A	None
Diff Pressure Transmitter	#16 AWG 5A MIN	45V DC Max	1/8 A	0.2A	None
Temp Transmitter RTD'S	#16 AWG 5A MIN	24V DC Max	Current Lim. Ckt.	0.02A	None
Vibration Monitors (Loose Parts)	#16 AWG 5A MIN	28V DC	Current Limiting Circuit	0.1A	None
Thermocouples	#16 AWG TC-Cable	Negligible (Millivolts)	Current Lim. Ckt.	Negligible (μ A)	None
E/P - Electro Pnematic Converter	#16 AWG 5A MIN	45V DC Max	Current Lim. Ckt.	0.2A	None
CEDM Position Indicator (Reed Switches)	#16 AWG 5A MIN	28V DC Max	Current Limiting Circuit	0.3A	None
Annunciators	#16 AWG 5A Min	125V DC	Current Lim. Ckt.	.005A	None
Fire Detection and Plant Security Systems	#16 AWG 5A Min	24V DC	Current Lim. Ckt.	0.2A	None

APPENDIX 3A*

SUMMARY OF RESULTS OF ENVIRONMENTAL QUALIFICATION TESTS

A. Fan Cooler Motor Unit Test

B. Differential Pressure Transmitter Test Report Abstract

C. Containment Electrical Cable Environmental Qualification Tests

D. Underground Electrical Cable Qualification for Service in Potentially Submerged Environment

E. Electrical Penetration Termination and Connector Qualification Tests

F. Qualification Test of Limitorque Valve Operator in a Simulated Reactor Containment Post Accident Steam Environment

* Pursuant to the requirements of IE Bulletin 79-01B a re-evaluation of the environmental qualification of electrical equipment installed in the plant was performed. This updates the information provided in this appendix. Complete documentation of the re-evaluation is available in the Records Vault at the site. See Section 3.11 for referencing to FPL responses to the bulletin.

Information provided in this appendix is historical and shall not be updated; however, it may still be similar to the re-evaluation documentation if no changes have taken place.

A. FAN COOLER MOTOR UNIT TEST

Tests reported in 1969 demonstrated the effectiveness of a heat exchanger assembly in isolating motor windings from the steam and chemistry environments of a post design basis event in a nuclear reactor containment. Subsequent tests have established long-term bearing life of such a motor and the radiation resistance of the windings. Additional steam exposure tests are now reported to comply with all provisions of the Guide for Type Tests of Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations, IEEE Guide 334. Also, these tests qualified design features not included in the original motor. Steam exposure tests of IEEE 334 were also performed on the same motor without the heat exchanger to qualify the motor for short-term, postevent operation. The Guide makes no provision for extrapolation to long-term postevent service and neither prescribed tests nor others conducted to this date are considered adequate to insure long-term, postaccident service without benefit of a heat exchanger.

B. DIFFERENTIAL PRESSURE TRANSMITTER TEST REPORT ABSTRACT

The following is an abstract of Fischer & Porter Co. Test Report DP-2204-51-B-006 dated December 2, 1968 which was issued as a qualifying type test for Fischer & Porter Co. pressure transmitters under special environmental conditions.

1. PURPOSE

The purpose of this project was to design a Differential Pressure Transmitter, which could withstand a special environmental application of live steam for a period of up to 24 hours. This special application was necessitated due to a failure of a pressure vessel or steam line in an enclosed housing, which contains an atomic reactor. The application of this instrument is for the nuclear power industry.

2. CONCLUSIONS

Engineering was successful in meeting the specifications. Using Figures 1 and 2 of this specification, which covers the steam temperature-time cycle, a unit was successfully run under these environmental conditions.

The results of these tests during the maximum steam and temperature cycle (90 psia saturated steam - 320°F) was a zero shift after 1 hour of +3.5% of span (see Figure 4). A calibration run was made at this time and no measurable span shift was observed. It should also be noted that during the entire time cycle a differential pressure of 30% of span was applied. At the end of an additional 1-3/4 hours at 60 psia saturated steam pressure, a calibration run was made with a resulting zero shift of +3% of span and no measurable span shift (see Figure 4).

At the end of the test, which was of 24 hours duration, a final calibration run was made after the unit had returned to ambient, and the resulting offset at zero was -1.5% of span (see Figure 6). There was no measureable change in span. It is felt that this final offset was due to the fact that this was the first time the unit had been subjected to such a high ambient temperature.

The pressure to current transducer (P/I), which will also be used in the same environmental conditions, will also meet the same specifications. The transmitter has the same basic components as those used in the electronic differential pressure transmitter, with the exception of the pressure elements. The electronic components, that is, the oscillator-amplifier, detector and force motor are identical. All modifications made to the electronic DP transmitter are also being made to the pressure to current transducer. The transmitter housing, which uses the same cover as the DP transmitter, will be sealed from the steam environmental conditions in the same manner as the electronic DP transmitter.

It should be noted that since the pressure element is in the sealed housing in the pressure to current transmitter, that during the temperature cycle there will be an increase in pressure in the housing of up to 10 psig. Since this pressure is on the outside of the pressure element it will have an effect of changing the zero. However, with the pressure ranges specified, the zero shift will be quite small. As an example, with a 1000 psi rated pressure the zero shift would be 1%, and this can be projected down to using a 100 psi element - the zero shift would be 10%. Therefore, it becomes essential that if the process pressures are low enough, and the zero shift would be detrimental to the operation of the unit, that absolute pressure transmitters should be specified and used.

To insure that the production transmitters, both P/I and DP, meet the specifications, two additional tests will be added to the standard production final testing. These tests will be as follows: a) All oscillator-amplifiers will be temperature tested at 320° F for a period of 1/2 hour to insure their functioning at the higher temperature; b) After final assembly and calibration the transmitters will be pressure checked, that is, 75 psig of helium will be added inside the transmitter housing to check for any possible leaks. The leak test will be conducted on the Mass-Spectrometer leak detecting device.

3. DISCUSSION

The following is a discussion of the methods and procedures used in testing the unit under the environmental conditions specified. A high pressure chamber capable of withstanding pressures of up to 500 psig was used to contain the DP transmitter being subjected to the test. The existing pressure line facilities at Fischer & Porter contained 100 psig steam pressure, and this was tapped into, and a special set-up made. A feeder line was connected with a separate steam pressure regulator, and from this a special high temperature flexible tubing was attached. A pressure gauge was set at the steam input to the test pressure chamber, and a valve and pressure gauge were attached at the output of the test pressure chamber. The valve on the output was regulated to maintain a back pressure of 75 psig, thus assuring a continuous flow of steam at all times during the test.

The output of the transmitter (4-20 milliamp) was brought out of the pressure vessel through special connections by use of high temperature insulated wire. This output was recorded on a 2202 series electronic recorder, and also monitored on a digital voltmeter. A record of this output is attached in the appendix. (see Figure 3 through Figure 6).

The 30% differential pressure was maintained by exposing the low pressure side to atmosphere through connections to the outside, and a 30% of span pressure applied to the high pressure side by a precision regulator, using a Press-I-Cell (.05% of span accuracy) to monitor this pressure. The range of the transmitter was set at 450" H_20 on a 150 to 1500" H_20 range transmitter, which is equivalent to 60" H_20 on a 20 to 200" H_20 unit. The only difference between these two units is that the high range unit uses a bellows as a sensing element and the lower range unit uses a diaphragm. All other components are identical.

During the course of this investigation we experience no problems under the temperature specification once the high temperature oscillator-amplifier was designed. We also experienced no problems due to the pressure that the instrument would see, that is 75 psig. The major problems occurred when the unit was subjected to the saturated steam whether at 25 psia or 90 psia. Because of this condition the additional modifications were necessary, including the sealing of the housing to prevent the steam from entering the transmitter. Thus it becomes essential that the cover be screwed tightly onto the base to prevent the steam from entering into the housing.

4. APPENDIX

The following data is attached:

a. Containment Environmental Conditions

Temperature vs. Time	Figure 1
Pressure vs. Time	Figure 2

b. Excerpts from the transmitted signal as recorded on the 2202 electronic recorder:

Room Temp. Calibration, 8 Oct.	Figure 3
Calibration runs at 90 psia & 75 psia; saturated steam	Figure 4
Calibration at 50 psia	Figure 5
Room Temp. Calibration, 9 Oct.	Figure 6

C. CONTAINMENT ELECTRICAL CABLE ENVIRONMENTAL QUALIFICATION TESTS

As stated in Section 3.11.3, tests have been performed on cable equivalent to designated categories I-A and I-B. Table 3A-I is a tabulation of design data and environmental test results which demonstrate that Category I-A and I-B cable purchased for use inside the containment will withstand their respective environmental design bases listed in Table 3.11-1.

CONTAINMENT ENVIRONMENTAL CONDITIONS

TEMPERATURE VS TIME

(NEED TO INSERT GRAPH)

CONTAINMENT ENVIRONMENTAL CONDITIONS

TEMPERATURE VS TIME

(NEED TO INSERT GRAPH)

(NEED TO INSERT FIGURE 3 & FIGURE 4)

(NEED TO INSERT FIGURE 5 & FIGURE 6)

THE
OKONITE
COMPANY

Post Office Box 340
Ramsey, New Jersey 07446
201-825-0300/Cable Okonite

RADIATION CERTIFICATION

Florida Power & Light Co.
Hutchinson Island Plant

The Okonite Company certifies that the cable presently on order under Ebasco Purchase Order No. NY 422274 and its supplement No. 1 will be serviceable for a period of one year or longer after a postulated reactor incident following forty (40) years of normal operation as described in Ebasco Specification Nos. FLO-8770-292-B, Rev. 3 of October 6, 1971 and FLO-8770-291-B, Revision 2 of August 16, 1971.

The environmental conditions listed in the specifications, as changed by your letter to us of December 15, 1971, are as follows:

The cable shall be designed to be serviceable for the following service cycles:

(a) 39 years and 11 months, normal conditions

Temperature	-	120° F (49°C)
Pressure	-	0 psig
Relative Humidity	-	40%
Radiation	-	1 Rad/hr

(b) Two Hour Emergency Environment Conditions. The cable shall be designed to be serviceable when subjected to the environmental conditions listed below after meeting the normal conditions in (a).

Temperature	-	270° F (130° C)
Pressure	-	44 psig
Relative Humidity	-	100%
Avg. Radiation Dose		2×10^6 Rads/hr.*

(c) Two to Twenty Four Hour Emergency Environment Conditions. The cable shall be designed to be serviceable when subjected to the environmental conditions listed below after meeting the conditions in (a) and (b).

Temperature	-	240° F)115° C)
Pressure	-	27 psig)
Relative Humidity	-	100%
Avg. Radiation Dose		1×10^6 Rad/hr.

(d) Maximum Long Term Emergency Environmental Conditions - Twenty-Four Hours to One Year. The cable shall be designed to be serviceable when subjected to the environmental conditions listed below after meeting the conditions in (a), (b) and (c).

(d) (continued)

Temperature	-	190° F (85° C) decaying to 150° F (66° C) at the end of 31 days and to 130° F at the end of one year
Pressure	-	25 psig decaying to 5 psig at the end of 31 days and to 1 psig at the end of one year
Relative Humidity	-	100%
Avg. Radiation Dose		150×10^3 Rads/hr. decaying to 3×10^3 Rad/hr. at end of 31 days and to 100 Rad/hr. after 90 days.

A borated water spray of approximately 1660 ppm as Boric Acid shall be applied continuously during the one year environmental conditions. (b), (c) and (d)

* The decay of radiation for the accident conditions were derived from the following:

2×10^6 Rads/hr., decay to 1/2 at 10 hours, 1/3 at 20 hours, 1/4 at 40 hours, then 100 Rads/hr. at 90 days.

The Okonite Company further certifies that tests have been performed on cables of the same formulation and construction, that validates this certification.

THE OKONITE COMPANY

F. M. McAvoy
Director - Cable Evaluatio

Sworn and subscribed before me this

third day of January, 1972

Doris C. Grader

Doris C. Grader, Notary Public of New Jersey
My Commission Expires
December 15, 1975

TABLE 3A-1

CONTAINMENT ELECTRICAL CABLE ENVIRONMENTAL QUALIFICATION TESTS

Vendor	Environ-mental Classifi-cation	Usage	Insulation Material	Jacket Material	Qualification Tests Performed					
					Accelerated Aging	Radiation Insulation	Resistance (D) Jacket	Post-LOCA Environment	Flame Tests	
Boston Ins Wire & Cable Co.	IA	Radiation* monitoring Sys	Silicone Rubber or XLPE	CSPE	Not Performed	1.8×10^8	1.8×10^8	Passed	B	IPCEA S-19-81 & S-66-524
	IB	Radiation monitoring Sys	PVC or XLPE	CSPE	Not Performed	1×10^6	1.8×10^8	Passed	B	IPCEA S-61-402 & S-66-524
Cerro Ins & Cable Co.	IA	Safety Related*' and Post LOCA Monitoring Equip.	XLPE	Neoprene	Not Performed	1×10^8	1×10^8	Passed	A,B	IPCEA S-66-524
	IB	General (See Note #2)	XLPE	Neoprene	Not Performed	1×10^8	1×10^8	Passed	A,B	IPCEA S-66-524
Okonite 5	IA	Containment* Fan Coolers	EPR	Neoprene	Not Performed	1×10^8	1×10^8	Passed	B,C	IPCEA S-68-516
Raychem Corp. & Cable Co.	IA	Safety Related* and Post LOCA Monitoring Equip.	XLPE	XL Polyo-lefin	In Progress	2×10^8	2×10^8	Passed	A, B	IPCEA S-66-524
	IB	General (See Note #2)	XLPE	XL Poly-olefin	In Progress	2×10^8	2×10^8	Passed	A, B	IPCEA S-66-524
Continental Wire & Cable Co.	IB	General	XLPE	PVC	Not Performed	1×10^8	1×10^7	Passed	B,C	IPCEA S-66-524 Plus ISA for Thermocouple
General Cable Co.	IB	General	XLPE	PVC	Not Performed	2 x 108	5 x 105	Passed	B,C	IPCEA S-66-525 S-61-402
Rome Cable	IB	General	XLPE	PVC	Not Performed	5×10^8	1×10^8	Passed	B,C	IPCEA S-66-524

Notes

1. Above cables consist of manufacturer's standard materials.
2. 1A & 1B cables are identical for these vendors but purchased under different specifications with each cable type having it's own unique bill of material number.
3. All new replacement 600V power and signal cable will be kerite and okonite brand cable. Kerite and okonite insulation maintains the dry/wet/alternately wet and dry properties of crosslinked polyethylene cable (CLPE or XLPE) while offering greatly enhanced fire retardency capability. See references to Section 8.3, References I and 2. Environmental qualification data can be found therein.

Symbols

* Inside Containment only
A IEEE Vertical Tray Flame Test
B IPCEA S-19-18 Section 6.19.6 or equal
C UL-44 horizontal
D Units in RADS

D. <u>UNDERGROUND CABLE QUALIFICATION FOR SERVICE IN POTENTIALLY-SUBMERGED ENVIRONMENT</u>

All plant cables, including those for Class 1E service, are suitable for service in wet or dry locations and are tested in accordance with Insulated Power Cable Engineers Association, (IPCEA), National Electrical Manufacturers Association (NEMA), Association of Edison Iluminating Companies (AEIC) Standards, and purchase order specifications (as applicable) to insure their suitability for normal, (dry, alternately wet and dry, and submerged) conditions of service in tray, conduit and underground installations.

Class 1E cables are not directly buried or installed for submarine service. These cables are installed indoors in cable tray or conduit and outdoors in underground duct banks. The underground duct banks are installed above the normal water table.

Concern has been expressed by the Regulatory Staff as to the compatability of St. Lucie underground cable with the service environment, i.e., have the cables been adequately qualified for the conditions they might experience during their service life. The propriety of utilizing the cables for this service has been reevaluated and the suitability for the service environment reaffirmed. Basically, the environmental qualification study consisted of three parts, namely,

1. Insitu Cable Experience

   The underground cable system of Miami Beach closely duplicates the St. Lucie service environment. The performance of cable on this system was reviewed to determine whether or not the dry, alternately wet and dry, and flooded conditions experienced by the Miami Beach cable results in cable failure due to deterioration. This review, which included all failure data since 1945, indicates that there has "never" been an electrical deterioration failure of any kind in the Miami Beach system.

2. Experimental Confirmation

   Four cable manufacturers have performed independent tests to demonstrate that the modern insulations utilized for St. Lucie cable have superior resistance to electrical deterioration than the insulations that have performed so well at Miami Beach. Each cable manufacturer has independently demonstrated that the modern insulations are superior. Thus, the St. Lucie cable life will undoubtedly exceed that of the Miami Beach cable.

3. Expert Opinion 11

   Two nationally recognized cable experts have reviewed the cable for the intended service environment. Both conclude that it is adequately qualified for the service environment.

The qualifications of the cable experts are provided as Tables 3A-D1 and 3A-D2. Details of the Miami Beach service experience and the testing conducted by the cable manufacturers is provided infra. The conclusion reached is that the cables have been adequately qualified for the service environment. In this regard it is noteworthy that the new insulations exhibit stable electrical properties (power factor), i.e., test data indicate that the propensity of the modern cable is to stabilized electrical properties. Because of this,electrical stability tests in 90°C water were terminated after 36 months.

Insitu Cable Experience

Confidence in the ability of the underground cables at St. Lucie to operate satisfactorily in a potentially submerged or alternately wet and dry environment can be demonstrated by extensive Florida Power and Light experience with underground duct and manhole installations in which cables have actually been subjected to environmental conditions similar to or more severe than those that can be expected at St. Lucie. The Miami Beach underground electrical distribution system installation provides an excellent experience base to establish a confidence level for the St. Lucie installation. The bases being that the Miami Beach system provides;

a) An extensive and long established underground distribution system with a systemized recorded failure data base for the last thirty years.

b) A natural environment which includes high temperatures, decaying vegetation, torrential rains, and hurricanes.

c) An underground distribution system that sees the potential wet and dry conditions, or combination thereof, that are characteristic of the St. Lucie installation. Part of the system is continually submerged; part is installed near the normal tide line and hence, is wet twice a day; and the remainder is installed above the normal high tide line and hence is subjected to occassional submergence from rain or hurricane driven sea water.

It is noteworthy that in the Miami Beach system no attempt has ever been made to control the water level in any underground installation except as required to facilitate personnel access into the underground system and manholes. In addition one of the cable experts, Mr. W. A. Thue has been personally involved in the Miami Beach system. Being directly involved in system development and normal occurrence evaluation for 25 years.

Table 3A-D4 provides a chronology of 600 volt cable installation on Miami Beach by type since 1957. Table 3A-D3 provides the cable feet - years experience record for this cable since 1957. Data on cable feet prior to 1957 is not readily available. However, cable failure records since 1945 are available, thus any correlation of failure data and cable feet - years based on Miami Beach data would be conservatively biased.

Since 1956 Florida Power and Light has installed over 600,000 feet of cable ranging in size from #12 AWG to 1500 kcmil butyl rubber insulated 600 volt rated cable on Miami Beach. This represents over 7,000,000 cable feet-years of experience. Many more cable feet of cable were installed prior to 1956. Over the facilities service life the St. Lucie underground installation represents but a small fraction of this greater than 7,000,000 cable feet years insitu experience base.

A review has been completed of all cable failure records since 1945 along with confirmatory interviews of supervisors, splicers, and engineers in the Miami Beach district. The results indicate an extraordinary experience record. There has "never" been an electrical deterioration failure of any kind on these cables or the splices associated with them. It is also noteworthy that there is no known data to contradict this outstanding reliability record from any other utility that has utilized butyl rubber insulated cable in underground installations. It must be noted however, that failures do occur. These have occured due to mechanical damage (e.g. damage during installation) that are not germane to a discussion of environmental qualification of cables. Preservice testing procedures at St. Lucie detect the presence of such mechanical damage prior to plant operation, thereby obviating the need to consider these failures during inservice operation.

Experimental Confirmation

Butyl rubber insulated cables have had excellent in-service experience as indicated by the Miami Beach underground distribution service record. However, technological advances has made available newer superior insulating materials, namely, crosslinked polyetheline (XLPE or CLPE) and ethylene propylene rubber (EPR or EPM). These newer insulations have been utilized widely by the industry for the last five to ten years. They provide superior performance to that experienced with butyl rubber.

Since an acceptable inservice data base is available for the butyl rubber insulation, qualification of the newer materials is readily accomplished through laboratory testing. The new materials and the butyl rubber must undergo appropriate identical tests. To qualify the new materials it is necessary and sufficient that these materials have resistance to electrical deterioration equal to or greater than butyl rubber. The test results clearly indicate the superior characteristics of these new materials. Whatever tests these newer insulations are subjected to as a means to determine their electrical stability in wet or dry locations, the new crosslinked polyethylene or ethylene propylene materials out perform the old butyl types Florida Power and Light has in service on Miami Beach by 2 to 6 times or more.

Typical evidence in regards to the superior performance to be expected with crosslinked polyethylene or ethylene propelene rubber is provided as follows;

1. Excerpts concerning moisture resistance from technical paper, T 74 044-4 presented by Okonite at the IEEE Power Engineering Society Winter Meeting in 1974. (Attachment 1 to Section 3A Part D)

2. The Okonite Company (Attachment 2 to Section 3A Part D)

   a) An Okonite Company letter of November 27, 1974 to Ebasco Services, Inc.

   b) Graphical representation indicating stability of various insulations in water at a temperature of 90°C.

3. General Cable Corporation (Attachment 3 to Section 3A Part D)

   a) Test results of long term immersion in water at 90°C.

   b) Aging evaluation of General Cable for inside and outside containment.

4. Cyprus Wire and Cable Company (formerly Rome Cable) (Attachment 4 to Section 3A Part D)

   a) Cyprus Wire and Cable Company letter of November 26, 1974 to Ebasco Services, Inc.

5. Raychem Corporation (Attachment 5 to Section 3A Part D)

   a) Raychem Corporation letter of November 27, 1974 to Ebasco Services, Inc.

The following provides a summary of the cable tests and results thereof:

1. Moisture resistance (See Attachment 1)

   In the 1950-57 era IPCEA developed a 16 week test procedure based on a continuous immersion at 50°C while under 600 volts dc to provide a means of assessing the effect of moisture resistance on cable life. Today, modern insulations can be immersed at 75°C, under the same dc potential, for 1 1/2 to 2 years, or more. Cables at the first generation reactors (1957 vintage), i.e. Shippingsport, Indian Point and Peach Bottom, have not experienced insitu problems due to moisture. This experience adds to the large base of successful cable performance, whereas, the test data provided on modern insulations reaffirms the high confidence level associated with these new materials.

2. The Okonite Company (See Attachment 2)

   The Okonite Company compared the performance of the older butyl insulation, used so successfully in the Miami Beach distribution system, with the performance of the crosslinked polyethylene and ethylene propelene rubber insulation, used on the cables installed at St. Lucie. The tests were made on small sample cables at 90°C, which is the maximum operating temperature for the insulations. Power factor, which is a measure of cable losses, is used to indicate insulation integrity, and is one of the best methods to determine degradation due to moisture.

   The results indicate that at the start of the test the butyl insulation has a power factor of about 5 percent. (See Figure 3A-D1.) After 12 months the power factor has increased to 30 percent at which point the insulation is judged to be unfit for service. In contrast, the "Natural CLPE", the insulation used at St. Lucie Unit 1, has an initial power factor less than one percent. As the test continued, the power factor decreased slightly to about one half of one percent. After 36 months the test was discontinued due to the remarkable electrical stability of the insulation.

The tests show that for EPM insulation (more generally called EPR) the initial power factor was 2.7 percent which decreased to about two percent and remained stable. Again the test was discontinued at 36 months.

These tests indicate that the crosslinked polyethlene and EPR insulations have lower initial power factors than the butyl insulation. Both the crosslinked polyethylene and the EPR insulations performed without deterioration for 36 months. With the butyl insulation, the power factor increased continually and reached an intolerable value within 12 months. In conclusion, the crosslinked polyethylene and EPR insulations show superior performance for at least three times as long as the butyl insulation, which gave satisfactory service in the Miami Beach distribution system.

3. General Cable Corporation (See Attachment 3)

   A test performed by General Cable provides an indication of long life expected for crosslinked polyethylene insulation. The cable samples were energized continuously at 600 volts. The life of the sample cable immersed in 90°C water was 951 days or approximately 2.6 years. It must be noted that failure occurred At a test overvoltage of 1200 volts a-c. This is well above the operating voltage (480V) for this class of cable at St. Lucie.

   An evaluation was performed and documented by Florida Power & Light Power Plant Engineering (EPO-86-805-E-2). Utilizing the LOCA profile of the Franklin Institute Test certified by General Cable for the cable supplied for St Lucie Unit 1 by their letter of May 3, 1973 (found in St Lucie Unit 1 Document Package 8770.A.451 8.0, Section 3), it can be demonstrated that the cable is qualified for 40 years of containment service plus the required Design Basis Event (DBE) service. The same documentation also demonstrates that the cable is qualified for 40 years of Steam Trestle service plus the required Design Basis Event (DBE) service.

4. Cyprus Wire and Cable Company (Formerly Rome Cable) (See Attachment 4)

   Air oven tests were performed by Cyprus on crosslinked polyethylene insulation. These tests were made at 135°C, 150°C and 175°C. The Arrhenius plot of these points indicated life of 50 years at 90°C, the rated maximum insulation operating temperature. (See Figure 3A-D2).

   Air oven tests and Arrhenius plots represent one method used by cable manufacturers to evaluate cable insulations. These tests supplement tests in water, examples of which have been previously discussed. The objective is a balance of electrical and physical properties which will insure long life under operating conditions.

5. Raychem Corporation (Attachment 5)

   Raychem has tested sample cables continuously immersed in water at 75°C in excess of twenty months with no failures. This data confirms the test results obtained by the other cable vendors.

Conclusion

In summary, the past performance of similar cables in an even more severe environment has shown their service reliability to be unaffected by any form of water for over 25 years. The newer insulations have been evaluated All accelerated tests indicate that they outperform those cables that have been used at Miami Beach.

The foregoing experience and testing demonstrate ipso facto that the cables installed at St. Lucie will have greater than a 40 year service life in a dry, alternately wet and dry, or potentially submerged environment.

TABLE 3A-D1

QUALIFICATIONS

W. A. Thue

William A. Thue is the System Operations Engineer - Underground for Florida Power & Light Company. He has been employed there since 1946 in various positions in the Engineering, Construction and Operating Departments. His present assignment involves the responsibility for all underground transmission, distribution and power plant cables and associated equipment as staff to the Group Vice President of Operations.

He is presently Vice Chairman of the Insulated Conductors Committee of the Power Engineering Society of the Institute of Electrical and Electronic Engineers. He is the immediate Past Chairman of the the Cable Engineering Section of the Association of Edison Illuminating Companies which is the source of cable standards for all paper insulated cables in the United States. He also serves as their Chairman of Extruded Dielectric Cable Standards. In this field, he also serves as the Chairman of the Joint Association of Edison Illuminating Companies-Insulated Power Cable Engineers Association Committee for Extruded Dielectric Power Cables.

He is a member of the U. S. National Committee for High Voltage Cables of CICRE (International Conference of Large Electric Systems), Committee C-8 (Electric Cables) of American National Standards Institute, of the Electric Power Research Institute's Research Project 78 for High Voltage Cables, and the Task Force for Power Cable Ampacities.

He is the author or coauthor of technical papers such as the "Shielding Performance of Power Cables", "Thermal Backfill for Transmission Cables", and "Improved Low Voltage-Direct Buried Cables".

TABLE 3A-D2

QUALIFICATIONS

L. D. Cronin

Cable Specialist for Ebasco Services Incorporated, New York, New York. Twenty-nine years of experience with this company.

Director for 12 years of Ebasco Underground Distribution Research Program that reviews continuously the state of the art in Underground Distribution.

Fellow of Institute of Electrical and Electronic Engineers Power Engineering Society.

Consultant for Underground Power Transmission Cable Installations in United States and in Europe and Asia.

Consultant for Underwater Power Transmission Cable Installations in United States and in Asia.

Advisory Member of Association of Edison Illuminating Companies Cable. Engineering Section that prepares National Cable Specifications. Chairman of Task Groups on

Guides for Cable Temperature Limits
Cable System Test Voltages

Member of Institute of Electrical and Electronic Engineers Power Engineering. Society Insulated Conductors Committee that prepares Guides for Cable Installation and Operation. Chairman of Overseas Practices Subcommittee.

Edison Electric Institute Chairman of Joint EEI-Bell Systems Subcommittee to Study Buried Distribution Systems that prepares Guides for Installation and Operation of Underground Joint Power and Telephone Installations.

Member of International Electrochemical Commission that prepares International Cable Standards. Member of

Technical Committee on High Voltage Cables
Technical Committee on Low Voltage Cables

Served as Member of Federal Power Commission

Advisory Committee on Underground Power Transmission
Advisory Committee on Power Distribution

TABLE 3A-D3

600 VOLT CABLE EXPERIENCE SUMMARY

MIAMI BEACH

Year	Total 600v Cable Feet	No. of Years	600V Cable Ft.-Yrs.
1957	81087	17	1378479
58	55493	16	887888
59	58887	15	883305
60	12207	14	170898
61	42144	13	547872
62	37156	12	445872
63	32851	11	361361
64	79822	10	798220
65	79571	9	716139
66	46411	8	371288
67	65118	7	455826
68	35952	6	215712
69	2328	5	11640
70	1347	4	5388
71	6608	3	19824
72	4139	2	8278
Total	641121		7277990

TABLE 3A-D4

600 VOLT CABLE RELEASED FOR INSTALLATION

MIAMI BEACH, FLORIDA

Description 600V Butyl Rubber Cable	1957	1958	1959	1960	1961
#12 AWG	6665	3954	4668	760	2550
6 AWG	19629	10822	11169	3250	10130
2 AWG	6713	4940	8785	1955	3039
4/0 AWG	29840	23297	16327	3158	19845
500 kcmil	13865	10048	12728	1716	5068
750 kcmil	3930	1491	3033	286	1286
1500 kcmil	445	941	2177	1082	226
Total	81087	55493	58887	12207	42144

	1962	1963	1964	1965	1966
#12 AWG	2210	1721	1910	840	1500
6 AWG	6205	14587	17510	34590	8340
2 AWG	5197	4905	7050	9112	6463
4/0 AWG	11768	8336	40377	26405	22195
500 kcmil	10210	2213	9722	5637	4422
750 kcmil	1427	842	2291	2560	3082
1500 kcmil	139	247	962	427	409
Total	37156	32851	79822	79571	46411

	1967	1968	1969	1970	1971	1972
#12 AWG	2480	-	-	-	-	1000
6 AWG	19314	3823	-	-		-
2 AWG	10670	9530	-	-	-	-
4/0 AWG	23256	17061	-	-	5538	3139
500 kcmil	5517	3472	1653	1300	-	-
750 kcmil	3387	1765	621	-	-	-
1500 kcmil	494	301	54	47	1070	-
Total	65118	35952	2328	1347	6608	4139

ATTACHMENT 1 TO SECTION 3A, PART D

Excerpt from paper T74 044-4, "Class 1E Cables for Nuclear Power Generating Stations", by E. E. McIlveen, V. L. Garrison, G. T. Dobrowski.

Moisture Resistance

Moisture resistance is a major factor in determining the normal life of a solid dielectric insulated conductor. It has become traditional to gain assurance of long life performance by totally immersing a #12 or 14 conductor insulated with a 45 mil wall of dielectric in water at an elevated temperature to accelerate the deteriorating effects of moisture. Monitoring the electrical properties then provides an indication of long term behavior. In the 1950-57 era with service gained experience that negative dc potential presented the most severe condition, IPCEA developed a 16 week test procedure along these lines based on a continuous immersion at 50°C while under 600 volts dc. At this time, more than sixteen years later, new generation moisture resisting insulations of similar geometry can be continuously immersed at 750°C while under the same dc potential, and survive from 1-1/2 to 2 years, or more. This is at least 5 times longer and at an effective temperature acceleration rate of 6 times greater than anticipated by the IPCEA procedure. Since insulated conductors of the 1957 vintage dielectrics installed at Shippingsport, Indian Point and Peach Bottom, among others, have not experienced distress due to moisture, it can be reasoned that control cable insulations now specified which have the capability of withstanding total immersion at 75°C under 600 V dc as discussed herein should develop the designed life of the cable plant. Fig. 1 presents data for a 45 mil wall of an ethylene-propylene base insulation conductor, and Fig. 2 illustrates the electrical behavior of a composite wall composed of 30 mils EP base plus 15 mils neoprene compound.

Reference to Table I discloses similar data for an ethylene-propylene base dielectric and also a flame resistant cross-linked polyethylene compound (FR-CLPE), but at 90°C continuous water immersion while under 600 V ac potential except when percent power factor (% PF) and the specific inductive capacity (SIC) are being measured at 40 and 80 V/mil ac. Following each test measurement the specimens were subjected to a 5 minute withstand test at 110 V/mil. The specific insulation resistance (SIR) were made at 500 V dc while at 90°C. The difficulty of predicting long term performance based on the customary 2 week test data is obvious. It may be of interest that the time to failure for a particular specimen is a complex function of several variables, one of which is the degree of mechanical perfection of the dielectric wall. Failure is often sudden with little or no forewarning, and occurs when the cable is undergoing 60 cycle power factor and capacity measurement, or during the subsequent withstand at 110 V/mil.

Fig. 3 not only shows the SIC values for an ethylene-propylene base insulation during a long term continuous water immersion study, but also the accelerating effect of temperature as manifested by a change in the 60 cycle capacity. The 142°C/42 psig steam autoclave exposure further accelerates tire increase in the SIC value but could change the reaction mechanism. In any event, if plotted on Fig. 3 the end point is still some two years out on the time scale.

(Need to insert Graphs)

ATTACHMENT 2 TO
APPENDIX 3A, PART D.

Post Ofice Box 340
Ramsey, New Jersey 07446
201-825-0300/Cable Okonite

THE
OKONITE
COMPANY

November 27, 1974

Mr. L. D. Cronin
Ebasco Services, Inc.
2 Rector Place
New York, New York 10006

Dear Mr. Cronin:

Subject: St. Lucie Plant -- Cable Performance Certification

With reference to our conversation on November 26, 1974, this was relative to Qualification Testing of Class IE Cables for Submerged Service, Ebasco Project No. FLO-8770.292L, RO-May 20, 1974 and in particular, Table I, sample #1 and #11, both single conductor 500 MCM cables.

We submit as evidence of suitability of samples #1 and #11 for submerged service the data presented in IEEE paper T 74 044-4, Table I and identified as FR-CLPE. The use of a 45 mil wall specimen as shown in Table I (instead of 110), the use of a 90°C bath (instead of 40), and the excellent performance during an 18 months immersion (instead of 1) are three major parameters which significantly accelerate the "life" simulation well beyond those called out in the referenced document. In addition, it may be noted that the sample in Table I had no external covering.

As further evidence, we submit data which may be found in IEEE paper 68 TP 651-PWR, Table IX, under sample CB-CLPE or NF-CLPE. This shows that whether the samples had been irradiated or not, they maintained a voltage withstand level at 80 V/mil for more than 32 days in a steam autoclave at 142°C (40 psig).

Relative to a discussion with Mr. William Thue, you will find enclosed a graph identified as Fig. 5. It may be noted that a butyl insulation which was identical to that which has given excellent service in "submerged service" reached an end point in accelerated immersion tests at 90°C in 12 months whereas the CLPE (natural) was still doing fine after 36 months, a factor of at least three times.

Very truly yours,

THE OKONITE COMPANY

E. E. McIlveen

E. E. McIlveen
Vice President - Engineering

EEM/row
cc: Mr. W. Thue

Attachments: T 74 044-4
68 TP 651 PWR
Fig. 5

The Okonite Company

(Insert Electrical Stability in 90c Water 14 awg wire, 0.47 wall)

Figure 3A-D1

ATTACHMENT 3 TO

APPENDIX 3A, PART D.

GENERAL
CABLE
CORPORATION

800 RAHWAY AVENUE, UNION, N.J. 07083 / (201)687-0250

December 17, 1974

Ebasco
Specification 211-69
FLO-8770-292-A

Florida Power and Light Company
Hutchinson Island Plant
Order No. NY 422273

Gentlemen:

In accordance with your recent request, we wish to advise the following information:

(1) The crosslinked polyethylene insulation employed by General Cable on cable fabricated for subject plant under referenced order exhibits the following long term electrical stability characteristics in water:

<u>Test Results Long Time Immersion In Water</u>
Water Temperature 90°C
600 Volts A.C. Applied between readings
Sample length 10 feet
14 AWG solid 30 mil wall XLPE

No. of Days Immersion	% Power Factor		Specific Inductive Capacity		Insulation Resistance Megohms/1000 ft. (3)
	(1)	(2)	(1)	(2)	
1	.39	.51	4.06	4.06	800
7	.39	.53	4.08	4.09	
14	.37	.53	4.08	4.08	
28	.42	.58	4.23	4.35	
54	.45	.61	3.94	3.94	
88	.36	-	4.07	-	
107	.39		4.12		
116	.41		4.14		
148	.48		4.24		
184	.59		4.27		55.4
230	.66		4.45		40.5
287	1.10		4.55		49.0
329	1.50		4.70		33.0
429	2.14		4.72		40.0
522	2.18		4.84		
612	2.64		4.79		30.0
666	2.64		4.78		30.0
697	2.31		4.78		50.0
738	2.12		4.75		55.0
764	2.27		4.77		60.0
814	3.47		4.79		25.0
874	2.76		4.82		32.0
951	*	-	*	-	*

(1) Measured at 40 volts/mil 60Hz
(2) Measured at 80 volts/mil 60Hz
(3) Measured at 90°C
* Failed at 1200 VAC 60Hz

The data indicates that the insulation has excellent electrical stability in water when exposed to the accelerated test temperature of 90°C. The performance shown would be further enhanced by the overall covering of polyvinyl chloride (individual jacket and/or overall jacket) stipulated in the specification.

(2) A "Certified Test Report" dated May 31, 1973 attached, provides further indication of the insulation and jacket performance under various environmental conditions involving moisture exposure.

(3) Relative to the performance of lead sheathing in a saline water environment, we refer you to "Corrosion of Metals II Lead and Lead Alloy Cable Sheathing" by R. M. Burns - Bell System Technical Journal. We believe pages 617 and 618 extracted therefrom, and attached, give clear indication of the adequacy of lead sheathing performance when exposed to a sea water environment.

We trust the information given herein is adequate to answer the question of cable performance capability raised by Mr. Dennis Cronin of Ebasco. The question related to ingress of salt laden water into the duct system in which certain cables supplied under FLO-8770 are installed.

Very truly yours,

David A. Silver,
Director of Engineering
Power & Control Operation

DAS:gch

ATTACHMENT 4 TO

APPENDIX 3A, PART D.

Cyprus Wire & Cable Company
421 Ridge Street
Rome, New York 13440
Telephone 315) 337-3000
Post Office Box 71
TWX 510)243-9732

Mr. L. D. Cronin
Ebasco, Inc.
2 Rector Street
New York, N.Y. 10006

November 26, 1974

Dear Mr. Cronin:

This is to certify that the 7 conductor #10 AWG XLP insulated PVC jacketed control cable for Florida Power & Light (N.Y. Order #422-338) is good for use in both dry and wet locations. This cable employs a UL RHH-RHW and XHHW insulation with a UL THW jacket. The following data should helpconfirm the suitability of this cable for use in wet locations:

C-51801-Rome 0-600 Volt XLP Insulation for UL RHH, RHW or USE and XHHW

3-Year Immersion in 75°C Water		
% Increase in Capacitance	-	5.10
Stability Factor	-	.13
IR in Megohms-1000 Feet	-	1000

Attached find an arrhenius plot which demonstrates the expected life of C-51801 XLP at its operating temperature.

Very truly yours,

S. R. D'Agostino
Methods & Standards

SRD:bes

Attachment

Insert the following:
Cyprus Wire & Cable co.
C51801
50% Residual Elongation
Figure 3A-D2

ATTACHMENT 5 TO

APPENDIX 3A, PART D.

RAYCHEM

November 27, 1974

Mr. L. D. Cronin
Electric Systems Consultant
Ebasco Services, Inc.
No. 2 Rector Street
New York, N.Y. 10006

Re: Ebasco Purchase Order NY422358
Raychem Bid No. 207
Florida Electric and Power Company

Dear Mr. Cronin:

This letter is in response to your inquiry regarding the assurances that we can give that the cables furnished under this order are suitable for operation in wet locations continuously, which can include operation submerged in water.

The instrument cables furnished are insulated and jacketed with Raychem Flamtrol^{TM}, which is of the generic class XLPE. This material has been designed for use in wet locations and has successfully passed the standard industry requirement for wet location service, which is to submerge a section of insulated wire in a water bath at 75°C and apply a negative d-c potential of 600 volts for 16 weeks. At two week intervals, a Dielectric Withstand Test is performed.

We have samples of Flamtrol that have been continuously immersed in water at 75°C for in excess of 20 months, with 600 volts of negative d-c applied to the conductor and which are periodically tested with an a-c withstand voltage. We also have measured the shift in capacitance over this period of time, as well as the stability factor. Within a few months, the capacitance had come to equilibrium value of approximately +12% and the stability factor is about 1.5. Both of these values are determined in accordance with the EM-60 Method.

Raychem Corporation
300 Constitution Drive
Menlo Park, California 94025
415/329-3333 TWX910-373-1778

The coaxial cables on this order are jacketed with the same material as the instrument cables, that is, Flamtrol. The dielectric material utilizes the same type of base resins as the jacket; therefore, these cables will perform in a wet location or submerged, as well as the Flamtrol instrument cables.

This should answer the questions that are being posed to you. Please call me again if I can help.

Very truly yours,

Frank E. La Fetra

Frank E. La Fetra
Market Manager - Utilities
Wire and Cable Division
415/329-3217

FEL/g

cc: Mr. J.A. Barresi - Raychem

Raychem Corporation
300 Constitution Drive
Menlo Park, California 94025
415/329-3333 TWX910-373-1778

E. ELECTRICAL PENETRATION TERMINATIONS AND CONNECTOR QUALIFICATIONS TESTS

Safety related containment electrical cable will be terminated at the penetration by splicing or where required for electrical shielding, with connectors. Qualification test results are:

a) Raychem Heat-Shrinkable Splicing Sleeves (WCFS-N)

The environmental qualification tests were in accordance with IEEE-383, "Proposed Guide for Type Tests of Class I Cables and Connections Installed Inside the Containment of Nuclear Power Generating Stations," and were found to withstand loss of coolant accident (LOCA) conditions either early or late in their anticipated useful life. The qualification method for complete splices was by means of "Type Tests" that considered pressure, temperature, radiation, chemical concentrations, humidity, and time. These tests exceeded the environmental design requirements applicable to the St. Lucie project.

New material was tested as well as heat aged material. The heat aging (121 C for 168 hours) was simultaneous with radiation exposure (cobalt 60 gamma radiation-200 x 10^6 rads total). All samples tested passed with no electrical or mechanical failures. A summary of the qualification tests is provided hereafter.

In addition, it should also be noted that the splicing sleeves have passed the following flame tests:

1) IPCEA Vertical Flame Test

2) IEEE Vertical Tray Test

3) UL224 Flame Test, FR-1 Rating

b) Connectors

Physical Science Nuclear connectors (Gulton Industries) have been qualified for post-LOCA conditions. These connectors or other environmentally qualified connectors will be provided. With regard to the Physical Science connector, it is manufactured from stainless steel and inorganic ceramic compounds. The following individual test results are listed to demonstrate the suitability of these connectors for the post-LOCA environment. No degradation was shown after a connector was exposed to 1 x 10^{13} rads (Gamma). Another connector was exposed to 450 F and 60 psi for 200 hours, and showed no signs of degradation. In addition, it should be noted that the integrity of the shell components has been shown by a helium leak rate test to be 2 x 10^{-8}cc per second and the insulation resistance of the mated connector is 1 x 10^{-13} ohms.

Summary of Raychem Qualification Tests

a) Introduction and.Abstract

This summary describes the evaluation of high voltage terminations and in line splices for use in nuclear power plant containments.

Using the Institute of Electrical and Electronics Engineers "Proposed Guide for Type Tests of Class I Cables and Connections Installed Inside the Containment of Nuclear Power Generating Stations" as the test basis, 5 and 15kv high voltage terminations (HVT's) and 600-2000 volt in-line splices (WCSF's) were found to withstand loss of coolant accident (LOCA) conditions either early or late in their anticipated use life. HVT's remain usable, having excellent tensile strengths and elongations, even after 200 Mrads (2 x 10^8 Rads) of gamma radiation in air. Properly applied HVT's form an environmental seal around the cable protecting it from high pressure steam, moisture, and boric acid spray.

b) Program Outline

HVT and WCSF evaluations were divided into two phases: materials evaluation and systems evaluation. The materials evaluation consisted of a look at how the materials of construction behaved as a result of nuclear radiation. Systems evaluation consisted of an analysis of how the completely-assembled parts behaved as a result of nuclear radiation and how well they withstood the effects of a loss-of-coolant accident before and after exposure to nuclear radiation.

The evaluation was based upon the Institute of Electrical and Electronics Engineers "Proposed Guide for Type Tests of Class I Cables and Connections Installed Inside the Containment of Nuclear Power Generating Stations." The test sequence for materials consisted of:

1) Heat aging the materials in a forced air oven at 121 $\pm$ 2 C for 168 hours.

2) Irradiation of the materials in a cobalt 60 gamma source at 0.52 Mrads per hour to total doses of 100 and 200 Mrads.

The test sequence for assembled high voltage terminations and in-line low voltage (i.e., 600-2000v) splices consisted of:

1) Heat aging high voltage terminated cables at 121 C $\pm$ 3 C for168 hours in a forced air oven.

2) Irradiation of assemblies with cobalt 60 gamma radiation at 0.50 Mrads per hour for HVT's and .27 Mrads per hour WCSF to total doses of 100 and 200 Mrads.

3) Subjecting irradiated assemblies, maintained at maximum rated voltage, to LOCA tests in a pressurized autoclave according to the following schedule:

a) 5 hours at 360 F, 70 psig steam.

b) 6 hours at 320 F, 70 psig steam.

c) 24 hours at 250 F, 21 psig steam, 0.2 percent boric acid spray, buffered to pH of 10.

d) 12 days at 221 F, 2.5 psig steam.

c) Tests Results

Tables 1 and 2 show the results of the materials evaluation of the HVT's and WCSF sleeves. These data show that even after 168 hours at 121 C in a forced air oven and subsequent irradiation to 200 Mrads cobalt 60 gamma radiation in air, the products have maintained a very high degree of mechanical integrity. As an example, the outer high voltage tubing and stress grading material have maintained at least 80 percent and 70 percent elongations, respectively. This coupled with the excellent tensile strengths, indicates that these materials have sufficient toughness and radiation resistance to withstand 200 Mrads gamma radiation.

Table 3 shows the electrical performance of 15kv HVT's during LOCA tests. From the data, it is evident that the HVT's are capable of performing during a loss-of-coolant accident. Applied voltages for the 15kv HVT's during LOCA tests varied between 8.7 and 15kv, phase to ground. Table 4 yields similar data for 5kv HVT's.

Table 4 yields similar data for 5kv HVT's. HVT's successfully withstand LOCA tests before and after irradiation. The 5kv HVT's were subjected to applied voltages between 5 to 8.6kv phase to ground during the LOCA sequence.

Table 5 shows electrical performance of a series of in-line splices (WCFS) made on 600, 1000 and 2000 volt class cable and subjected to continuous maximum cable rated voltages.

d) Conclusion

Data supplied in this summary show the high voltage terminations and in-line low voltage splices are acceptable for use in nuclear power plant containments. The assembled terminations and splices have successfully withstood LOCA tests and remain functional so as to permit safe and orderly operation of equipment under post LOCA conditions.

TABLE 1

EFFECTS OF NUCLEAR RADIATION UPON
RAYCHEM HVT MATERIALS

	Outer Tubing	Stress Grading
Initial elongation, %	260	236
Elongation after 168 hours at 121°C plus 100 Mrads, %	126	140
Elongation after 168 hours at 121°C plus 200 Mrads, %	80	70
Initial tensile strength, psi	2290	1560
Tensile strength after 168 hours at 121°C plus 100 Mrads, psi	3025	2015
Tensile strength after 168 hours at 121°C plus 200 Mrads, psi	3020	1665
Initial hardness, Shore D	43	37
Hardness after 168 hours at 121°C plus 100 Mrads, Shore D	57	50
Hardness after 168 hours at 121°C plus 200 Mrads, Shore D	60	50

TABLE 2

EFFECTS OF NUCLEAR RADIATION UPON
RAYCHEM WCSF MATERIALS

	WCSF Tubing Samples	WCSF Slab Sample @ .125” Thickness
Initial elongation, %	565	440
Elongation after 168 hours at 121°C plus 100 Mrads, %	---	145
*Elongation after 168 hours at 121°C plus 200 Mrads, %	100	70
Initial tensile strength, psi	2180	1600
Tensile strength after 168 hours at 121°C plus 100 Mrads, psi	--	1745
Tensile strength after 168 hours at 121°C plus 200 Mrads, psi	1500	1685
Initial hardness, Shore D	37	43
Hardness after 168 hours at 121°C plus 100 Mrads, Shore D	--	46
Hardness after 168 hours at 121°C plus 200 Mrads, Shore D	42	52

*Tubing samples were exposed to simultaneous heat aging and irradiation

TABLE 3

PERFORMANCE CHARACTERISTICS (CORONA EXTINCTION VOLTAGE) OF RAYCHEM 15KV HIGH VOLTAGE TERMINATIONS DURING DBE/LOCA TESTING[1]

	HVT #1	HVT #2	HVT #3	HVT #4	HVT #5	HVT #6
Initial CEV, KV	20	17.5	20	19.5	21.5	21
CEV after 168 hours at 121°C	30	25	25	24	31	19
CEV after 168 hours at 121°C plus 100 Mrads[2]	--	--	14	15.5	--	--
CEV after 168 hours at 121°C plus 200 Mrads[2]	--	--	--	--	19.5	16
CEV after 35 hours DBE[3]	17	15	20.5	21	16	16

Notes

1. Crosslinked polyethylene cable, copper tape shield, extruded semiconductive layer.
2. Cobalt 60 gamma radiation, dose rate of 0.50 Mrads per hour.
3. 5 hours at 360°F, 70 psig steam; 6 hours at 320°F, 70 psig steam, 24 hours at 250°F, 21 psig steam, 0.2% boric acid spray at pH of 10, 12 days at 221°F, 2.5 psig steam.

TABLE 4

PERFORMANCE CHARACTERISTICS (CORONA EXTINCTION VOLTAGE) OF RAYCHEM 5KV HIGH VOLTAGE TERMINATIONS DURING DBE/LOCA TESTING[1]

	HVT #1	HVT #2	HVT #3	HVT #4	HVT #5	HVT #6
Initial CEV, Kv	4.8	4.2	5.5	5.5	5.8	5.8
CEV after 168 hours at 121°C	4.5	4.3	4.6	4.0	4.8	4.4
CEV after 168 hours at 121°C plus 100 Mrads[2]	---	---	5.2	4.3	---	---
CEV after 168 hours at 121°C plus 200 Mrads[2]	---	---	---	---	4.2	5.0
CEV after 35 hours DBE[3]	4.5	4.1	4.7	4.5	4.8	(4)

Notes

1. EPR cable, copper tape shield, tape semiconductive layer.
2. Cobalt 60 gamma radiation, dose rate of 0.50 Mrads per hour.
3. 5 hours at 360°F, 70 psig steam; 6 hours at 320°F, 70 psig steam; 24 hours at 250°F, 21 psig steam, 0.2% boric acid spray at pH of 10, 12 days at 221°F, 2.5 psig steam.
4. Specimen mechanically damaged before being placed in autoclave.

TABLE 5

PERFORMANCE CHARACTERISTICS OF RAYCHEM
600-2000 VOLTS IN-LINE SPLICES TYPE WCSF
DURING DBE/LOCA TESTING

	New Material	Per Table 2 Aged 168 Hours @121°C & 200 Mrads
Electrical Strength 5 Samples-Volts/Mil		
Minimum	312	318
Maximum	491	355
$\overline{X}$	380	334
Wall Thickness	.084"	.086"
Volume Resistivity OHM-CMS	2.5×10^{13}	1.2×10^{14}
Flammability Per A.S.T.M. D-2863 Oxygen Index Note Slab Data Only	35.0	37.0

Notes

1. Cable types for testing
   A. 600 volt Flamtrol™
   B. 2000 volt EPR/Neoprene
2. All samples were continuously operated at maximum current and voltage per cable class. Current levels per I.P.C.E.A.
3. Cobalt gamma radiation, dose rate of 0.27 Mrads per hour.
4. 5 hours at 360°F, 70 psig steam; 6 hours at 320°F, 70 psig steam; 24 hours at 250°F, 21 psig steam, 0.2% boric acid spray at pH of 10, 12 days at 221°F, 2.5 psig steam.
5. All samples passed. No electrical or mechanical failures.

F. Summary of Qualification Test of Limitorque Valve Operators in a Simulated Reactor Containment Post Accident Steam Environment (F-C3441)

a) Introduction

Two Limitorque SMB-0-25 valve operators (prototype units for safety injection tank isolation valves and shutdown cooling suction line isolation valves) were subjected to a qualification test to determine their acceptability for service in the post-LOCA environment. The test consisted of a 30-day exposure to a steam environment at temperatures going as high as 340 F during the first day. The performance of the valve operators was monitored by periodic cycling (under simulated valve-seating load) and measurement of insulation resistance on all power and control leads. The test was started on July 31, 1972, and ran through August 30, 1972.

b) Identification of Valve Operators

The valve operators were identified by the following information on the name plates:

Unit No. 1
Limitorque Valve Operator

Type:	SMB
Size:	0
Order:	360943A
Serial:	144068

Motor

Manufacturer:	Reliance Electric Co.
Identification No:	601962-P
Start:	25 lb-ft
Run:	5 lb-ft
Type:	P
Frame:	R56
Phase:	3
RPM:	1700
Hz:	60
Volts:	230/460
Amp:	8.0/4.0
Rise at Run Torque:	75°C
Duty:	15 min
Insulation:	Class HR

Unit No. 2
Limitorque Valve Operator

Type:	SMB
Size:	0
Order:	355696A
Serial:	135809A

Motor

Identification:	463489-DX
Start:	25 lb-ft
Run:	5 lb-ft
Type:	P
Frame:	R56
Phase:	3
RPM:	1700
Hz:	60
Volts:	230-460
Amps:	8.0/4.0
Ambient:	75°C
Insulation:	Class HR
Duty:	15 min

The valve operator on Unit No. 1 had previously been exposed to gamma radiation (200 megarads) and a steam/chemical environment (for twelve days), and had been refitted by Limitorque with a motor which had been subjected to a gamma radiation dose of 200 megarads and a seismic test. The valve operator had also been subjected to a seismic test. Unit No. 2 had not been subjected to any prior testing.

During the installation of the units in the test chamber, the melamine switch base of Unit No. 1 was accidentally broken. It was replaced by a new base which was first exposed to 200 megarads of gamma radiation, the same radiation exposure which Unit 1 had received.

c) Test Procedure

The valve operators were exposed to steam in accordance with the pressure/temperature profile recommended in the proposed IEEE guide for type tests of Class I electric valve operators, Proposed Guide for Type Test of Class I Electric Valve Operators for Nuclear Power Generating Stations, Draft 13, IEEE Project No. 382, JCNPS/SC2.3, June 1972. This profile is illustrated in Figure 3A-1 which also shows the schedule for cycling the valve operators and measuring the insulation resistance of the power and control leads. During the first four days of the test, the specified temperatures and pressures were maintained by the controlled injection of steam into the test chamber. To achieve the specified temperature drops, the natural cooling of the test chamber (after the steam pressure and flow rate were decreased) was enhanced by blowing air over the exterior of the chamber and circulating water through a coil inside the chamber. During the remainder of the test, the 200 F/10 psig state was maintained by filling the test chamber with air and using external electrical heaters. The atmosphere within the chamber was kept saturated with water vapor by daily injections of steam and by maintaining the steam condensate in the bottom of the chamber at a temperature equal to, or slightly greater than that of the air/vapor mixture.

d) Test Results

1) Pressure/Temperature Profile

The actual pressure/temperature profile acheived during the critical first four days is illustrated in Figure 3A-2 During the last twenty-six days of the test, the temperature was maintained within approximately 5 percent of the specified 200 F. The main difference between the specified and actual temperature profiles is that more than the specified time was required to cool the test chamber after the first dwell at 340 F. (As a consequence of there being two valve operators within the test chamber, the amount of heat that had to be dissipated in two hours exceeded the cooling capacity.) Another difference is that the pressure rises to 105 psig required 19 seconds and 23 seconds at the beginning of the first and second pressure transients, respectively, instead of the specified 10 seconds. However, the temperature rose to about 330 F within 15 seconds at the start of the first transient and in 10 seconds at the start of the second transient. This was observed by viewing a thermometer at the top of the test chamber. The further rise to 340 F occurred more gradually.

Figure 3A-1 -- Specified Steam Exposure Profile
NEEDS TO BE INSERTED

Figure 3A-2 -- Actual Steam Exposure Profile
NEEDS TO BE INSERTED

2) Flooding of Test Chamber

During the fifth day of the test, when the conditions were being changed from 250 F/15 psig to 200 F/10 psig, it was discovered that the test chamber had become flooded with steam condensate. Investigation revealed that the condensate trap had become clogged with grease that had evidently come out of the pressure relief valve of the valve operators.

Judging by the amount of water which was removed from the chamber (about 225 gal) it was clear that the valve operators had been completely submerged by the condensate. This was confirmed at the conclusion of the test when the water line left inside the test chamber was found to be higher than the top of the valve operators under test. To help clear the interior of the valve operators of water which had entered them as a result of the flooding, air and nitrogen were flushed through the operator switch compartments (by use of the lines running between them and pressure gages outside the test chamber). Aside from this corrective action, the test was carried on without interruption; the test chamber was not opened at any time. The units performed normally when cycled after the period of flooding.

3) Operator Cycling Data

The electrical parameters (current, potential, and power) and the stroke times are listed in Tables 3A-1 and 3A-2 for Units 1 and 2, respectively.

The operators functioned normally throughout the test, with the following exception. Beginning with the third cycle, Unit 2 began to require two pushes of the start button to initiate the open cycle, after which the cycle was executed normally. An analysis of this effect and checks made after the valve operator was disassembled led to the following explanation.

At the start of the open cycle, the bypass functions to prevent the opening of the torque switch by the torque spring, which is released and bounces back when the open pushbutton is actuated. Evidently, although it functioned satisfactorily at room temperature, the bypass was not remaining active long enough to fulfill its function after the valve operator was heated to 340 F. This may have been due to a change in bypass setting resulted from the fact that the grease lubricating the spring became lighter when heated, and had less dampening effect on the spring bounce than it did at room temperature. Thus, it appears that two pushes of the start button were needed because of an improper initial setting of the gear limit bypass. It must be emphasized that, aside from the need for a second push of the start button, the open cycle was always executed normally.

TABLE 3A-1 VALVE ACTUATOR CYCLING DATA – UNIT NO. 1

Cycle No.	Time After Start of Test (hr)	OPEN: Potential ϕ ab (V)	OPEN: Potential ϕ ac (V)	OPEN: Potential ϕ bc (V)	OPEN: Running Current ϕ a (A)	OPEN: Running Current ϕ b (A)	OPEN: Running Current ϕ c (A)	OPEN: Power (W)	OPEN: Stroke Time (sec)	CLOSE: Running Current ϕ a (A)	CLOSE: Running Current ϕ b (A)	CLOSE: Running Current ϕ c (A)	CLOSE: Peak Current ϕ c (A)	CLOSE: Power Running (W)	CLOSE: Power Peak (W)	CLOSE: Stroke Time (sec)
1*	-0.58	480	480	480	3.6	3.3	3.6	250	74.0	3.6	3.5	3.4	3.6	250	2000	74.0
2	0.27	475	475	475	3.5	3.3	3.6	250	74.0	3.5	3.4	3.3	3.6	250	1000	75.0
3	2.0	475	475	475	3.4	3.3	3.5	250	74.0	3.6	3.4	3.3	3.8	250	1000	74.5
4	6.1	475	475	475	3.5	3.4	3.5	250	74.0	3.6	3.5	3.4	4.0	250	1000	74.5
5	8.1	475	475	475	3.6	3.5	3.6	250	74.0	3.7	3.6	3.5	4.0	250	1000	74.5
6	11.6	484	484	484	3.6	3.5	3.7	250	74.0	3.7	3.6	3.5	4.0	250	1000	75.0
7	101.9	478	478	478	3.5	3.3	3.6	250	74.0	3.6	3.4	3.3	3.8	250	1000	75.0
8	167.6	478	476	477	3.5	3.3	3.6	250	74.2	3.6	3.4	3.3	3.8	250	500	75.0
9	191.3	481	481	481	3.5	3.3	3.6	250	74.0	3.6	3.5	3.4	4.0	250	750	75.0
10	360.0	480	480	480	3.5	3.3	3.5	250	74.0	3.6	3.4	3.3	3.8	250	1250	75.0
11	575.6	478	478	478	3.5	3.3	3.6	250	74.0	3.6	3.4	3.3	3.9	250	1250	75.0
12	724.8	478	478	478	3.5	3.3	3.6	250	74.0	3.6	3.4	3.3	3.8	250	1250	75.0
13	726.7	480	479	480	3.6	3.3	3.6	250	74.0	3.6	3.4	3.3	3.8	250	1250	75.0

* Checkout cycle run before start of test.

TABLE 3A-2 VALVE ACTUATOR CYCLING DATA – UNIT NO. 2

Cycle No.	Time After Start of Test (hr)	OPEN: Potential ϕ ab (V)	OPEN: Potential ϕ ac (V)	OPEN: Potential ϕ bc (V)	OPEN: Running Current ϕ a (A)	OPEN: Running Current ϕ b (A)	OPEN: Running Current ϕ c (A)	OPEN: Power (W)	OPEN: Stroke Time (sec)	CLOSE: Running Current ϕ a (A)	CLOSE: Running Current ϕ b (A)	CLOSE: Running Current ϕ c (A)	CLOSE: Peak Current ϕ c (A)	CLOSE: Power Running (W)	CLOSE: Power Peak (W)	CLOSE: Stroke Time (sec)
1*	-0.35	480	480	480	3.2	3.4	3.5	250	75.5	3.4	3.6	3.3	3.8	250	1500	76.0
2	0.35	475	475	475	3.2	3.3	3.5	250	76.0	3.3	3.5	3.3	3.5	250	1250	76.5
3	2.2	475	475	475	3.3	3.4	3.5	250	--	3.3	3.5	3.2	3.7	250	1250	77.0
4	6.3	475	475	475	3.3	3.5	3.5	250	79.5	3.4	3.5	3.4	3.9	250	1000	76.5
5	8.2	475	475	475	3.4	3.5	3.6	250	76.5	3.5	3.6	3.4	3.8	250	1000	77.0
6	11.7	484	484	484	3.4	3.5	3.6	250	77.0	3.5	3.7	3.4	3.9	250	1250	77.0
7	101.0	473	473	473	3.2	3.4	3.4	250	76.5	3.4	3.6	3.3	3.8	250	--	76.0
8	167.8	475	475	475	3.2	3.4	3.4	250	75.5	3.4	3.5	3.2	3.6	250	750	76.5
9	191.2	481	481	481	3.3	3.5	3.6	250	75.5	3.4	3.6	3.3	3.9	250	1000	76.5
10	360.2	479	479	479	3.2	3.4	3.5	250	76.5	3.4	3.5	3.3	3.9	250	1250	76.5
11	575.5	478	478	478	3.2	3.4	3.4	250	76.0	3.3	3.5	3.2	3.8	250	1500	77.0
12	724.9	478	478	478	3.3	3.4	3.4	250	76.0	3.4	3.6	3.2	3.7	250	1250	77.0
13	726.6	480	481	480	3.3	3.4	3.5	250	76.0	3.4	3.6	3.2	3.8	250	1250	76.0

* Checkout cycle run before start of test.

4) Insulation Resistance Measurements

The measurements of insulation resistance made periodically on the power and control leads are listed for the two units in Tables 3A-3 and 3A-4. These measurements were made between each lead and ground (the test chamber), at 500 Vdc. The low values obtained on some of the control leads of Unit No. 1 during the first set of measurements (see first row of Table 3A-3 were believed to be caused by chemical deposits left on the switch components by prior testing. In the process of cleaning and reconnecting the affected parts, a switch plate was accidentally broken; and, as mentioned previously the new part with which it was replaced was first exposed to the same dose of nuclear radiation that Unit No. 1 had received in prior testing.

During the time that the units were flooded with steam condensate (see Section 3.3.2), the insulation resistances decreased significantly on both units; however, there was a recovery to high resistance values when the flooding was corrected.

5) Final Inspections

A visual inspection of the valve operators and test chamber was conducted at the conclusion of the test. The grease marks inside the test chamber clearly show that the condensate level rose well above the top of the valve operators during the flooding discussed above. This means that the motors were completely submerged during the flooding.

The paint had deteriorated over the entire exterior of the units, particularly on Unit No. 1 (which had gone through a steam/chemical exposure prior to the steam exposure reported herein); and corrosion and pitting of the metal surfaces had begun. No damage was apparent inside the switch compartments. On Unit No. 2, a white powdery material that seemed to be a foreign substance covered part of the melamine plate in the limit switch. The lower portions of both switch compartments were covered with a thin layer of grease that had become partially baked and was flaking in some areas. The interior of each switch compartment cover exhibited what seemed to be a water line about 3 to 4 inches from the top; evidently, air trapped inside the covers prevented the water from completely filling the compartments during flooding. However, the height of the water line was such as to indicate that all but the uppermost parts of the switch mechanism had been under water.

After the valve operators were returned to Limitorque, they were disassembled for more detailed inspection. On Unit No. 1, the gasket between the motor casing and valve-operator housing was in good condition. There was grease

TABLE 3A-3

INSULATION RESISTANCE OF POWER AND CONTROL LEADS - UNIT NO. 1

All Resistances are in Megohms Except Where a K Indicates Kilo-ohms

Time After Start of Test (hr)	←--------Stator Winding Leads----------→			←---------Control Circuit Leads---------→							
	T-1	T-2	T-3	CL1	41	45	51	55	61	70	71
-238.9*	>100	>100	>100	∞	0.6	0.6	0.7	0.7	∞	∞	∞
-65.9*	∞	∞	∞	∞	∞	∞	∞	∞	∞	∞	∞
-0.66*	500	500	500	200	∞	∞	∞	∞	∞	∞	∞
0.08	15K	15K	15K	50K	60K	70K	75K	75K	75K	75K	70K
1.97	70K	65K	65K	35K	35K	35K	40K	45K	42K	45K	42K
11.3	0.13	0.13	0.13	0.48	0.48	0.48	0.49	2.9	0.50	0.51	0.51
95.8	2.0K	3.0K	3.0K	1.0K	1.0K	1.0K	1.0K	2.0K	5.0K	5.0K	1.0K
99.3	5.0K	4.5K	4.5K	1.0K	1.0K	1.0K	1.0K	7.5K	5.0K	5.0K	1.0K
99.9	5.0K	4.5K	4.5K	-	-	-	-	-	-	-	-
101.9	7.5K	7.5K	7.5K	30K	30K	35K	35K	0.14	40K	40K	40K
167.1	∞	∞	∞	>100	>100	>100	>100	∞	>100	>100	>100
191.0	∞	∞	∞	>100	>100	>100	>100	∞	>100	>100	>100
359.8	<∞	<∞	<∞	50	50	50	50	∞	50	50	50
575.2	>100	>100	>100	40	40	40	40	∞	40	40	40
724.7	30	30	30	25	25	25	25	>100	25	25	25
726.5	∞	∞	∞	<∞	<∞	<∞	<∞	<∞	<∞	<∞	<∞

*Checkout readings taken before start of test,

TABLE 3A-4

INSULATION RESISTANCE OF POWER AND CONTROL LEADS - UNIT NO. 2

All Resistances are in Megohms Except Where a K Indicates Kilo-ohms

Time After Start of	←--------Stator Winding Leads----------→			←---Control Circuit Leads---→							
Test (hr)	2T-1	2T-2	2T-3	2CL1	241	245	251	255	261	270	271
-238.9*	∞	∞	∞	∞	∞	∞	∞	∞	∞	∞	∞
-65.9*	500	500	500	200	200	200	200	200	200	200	200
-0.66*	-	-	-	-	-	-	-	-	-	-	-
0.08	90K	80K	80K	60K	60K	60K	60K	60K	60K	60K	60K
1.97	0.13	0.13	0.14	39K	40K	43K	0.24	43K	48K	48K	46K
11.3	0.16	0.15	0.14	0.50	0.50	0.50	1.9	0.50	0.50	0.51	0.51
95.8	1.0K	l.5K	1.5K	1.0K	1.0K	1.0K	4.0K	1.0K	5.0K	5.0K	1.0K
99.3	8.0K	8.0K	8.0K	1.0K	1.0K	1.0K	4.0K	1.0K	5.0K	6.0K	2.0K
99.9	-	-	-	-	-	-	-	-	-	-	-
101.9	25K	25K	25K	40K	40K	40K	90K	40K	40K	50K	50K
167.1	90	90	90	>100	>100	>100	>100	>100	>100	>100	>100
191.0	90	90	90	>100	>100	>100	∞	>100	>100	>100	>100
359.8	8.5	8.5	8.5	50	50	50	100	50	50	50	50
575.2	7.0	7.0	7.0	40	40	40	90	40	40	40	40
724.7	6.5	6.5	6.5	25	25	25	80	25	25	25	25
726.5	<∞	<∞	<∞	<∞	<∞	<∞	<∞	<∞	<∞	<∞	<∞

*Checkout readings taken before start of test.

mixed with water on the pinion; there also was moisture on the inside of the motor end-caps, mixed with grease at the front end. However, the shaft and bearings turned freely, and lubrication seemed to be unimpaired. The drive gear was removed and found to be covered with grease, with no evidence of water. Water was found in the thrust tube at the valve end of the stem, but there was no corrosion. The grease at this location appeared to have broken down, possibly as a result of hydrolization or the exposure to nuclear radiation, but it appeared to have maintained lubrication of the stem. The tapered bearing on the drive sleeve was still well lubricated and there was no sign of wear. No moisture was evident on this bearing.

Unit No. 2 had much the same appearance as Unit No. 1, except that there was less evidence that there had been any breakdown of the grease in the thrust tube, possibly because Unit No. 2 had not been exposed to nuclear radiation.

While the observed partial breakdown of grease in the thrust tube is mentioned for completeness, this part of the unit was an attachment to permit simulation of the valve-seating load. In an actual installation, the external part of the stem (within the thrust tube in the test set-up) might not be lubricated at all.

e) Conclusion

Two Limitorque SMB-0-25 valve operators were subjected to a Qualification Test consisting of a 30-day exposure to a steam environment, including two temperature cycles going to 340 F during the first day. Unit No. 1 had previously been exposed to nuclear radiation, a seismic test and a steam/chemical environment. Both units were cycled periodically with a simulated valve-seating load during the test.

The pressure/temperature profile closely followed that recommended by a cognizant IEEE committee. The units were subjected to severe flooding with steam condensate during the first few days of the test; this happened because the condensate trap on the test chamber became clogged with grease that came out of the pressure relief valves of the valve actuators.

The units performed satisfactorily throughout the test in spite of the flooding. Inspection of the units following the test revealed that all parts were in satisfactory condition. It was evident that lubrication had been maintained in spite of the loss of grease. Although water had entered in some places, none of the internal parts were corroded.

APPENDIX 3B

SUMMARY OF RESULTS OF SEISMIC QUALIFICATION TESTS

A. 4.16 kv Switchgear

B. 480 Volt Switchgear and Station Service Transformers

C. 480 Volt Motor Control Centers (MCCs)

D. 125 Volt AC and DC Panels

E. ESF Actuation and Measurement Cabinets

F. Battery Charger

G. Station Auxiliary Motors

A. 4.16 kv SWITCHGEAR

The following is a summary of Westinghouse test report G.O. NYMI18414-YI dated February 1973. The report was issued as seismic qualification for the 4.16 kv switchgear.

PCM's 89350 and 89351 modified the 4.16 Kv switchgear to accept PIP as well as Monolithic style breakers. While the PCM states that the seismic qualification of the switchgear remains valid, NLI report R-037088-2, revision 1 documents in-cabinet response spectra which reflects the modified switchgear configuration.

1. Synopsis

A representative 50DHP350 metal-clad switchgear unit was subjected to an intensive test program per the intent of IEEE Standard #344 to qualify it for use in nuclear power stations with seismic vibration requirements. The seismic qualification of the 50DHP250 metal-clad switchgear unit is based on the test data of this 50DHP350 switchgear unit, a comparison of physical features, and static and dynamic analyses. The results of this test and analytical program have been analyzed in relation to the specific requirements of St. Lucie Unit No. 1 of The Florida Power and Light Company, per Spec. FLO-8770-284 with the floor response curves submitted by EBASCO Services Incorporated.

The conclusion is made that the test program and the subsequent correlation and additional analysis of special features of the subject switchgear has verified the ability of both the 50DHP250 and the 50DHP350 metal-clad switchgear furnished for this station to operate satisfactorily in the specified seismic environment.

2. Description of Equipment Tested

The 50DHP350 circuit breaker and a representative cell, complete with relays and auxiliary equipment required on most orders, were selected for testing. This breaker, which has an interrupting rating of 41,000 amperes at 4.76 kv, is one of the most common ratings found in nuclear power stations. The unit tested was a standard design or "off-the-line" unit with no attempt made to improve its seismic capability. The relays were selected to represent the various types of relay movements and were located on the front door by the same computer program normally used for this purpose on regular commercial orders.

The switchgear equipment was supported on three structural channels to duplicate normal service mounting. The structural channels, in turn, were welded to steel plates which were bolted to the vibration machine.

3. Test Results

The switchgear was tested independently in three directions: front-to-back, side-to-side, and vertical. In each direction, tests were made at the resonant frequencies found during the initial frequency sweep as well as at selected off-resonant frequencies. The resonant natural frequencies discovered in the test equipment are 7 Hz side-to-side and 9.5, 13, and 16 Hz front-to-back. In addition, the vibration magnification that occurs at resonance of the equipment during the continuous sweep frequency search permitted the determination of the critical damping factors in the switchgear.

The following damping factors at the various accelerometer locations are retabulated below:

Accelerometer Location	Side -to- Side	Front -to- Rear
Arc Chute	5%	7%
Door	-	5%
Cell	6.5%	13%
Side Panel	6%	-
Potential Transformer Support	-	10%

The lowest damping factor found in the switchgear was 5 percent. Therefore, 5 percent is a conservative figure for the qualification of switchgear equipment.

In all, over 75 tests were run at various peak amplitudes; each test consisted of 5 sine beats, simulating one earthquake; each sine beat contained 5 cycles of the specified vibration frequency. Tests were made with the breaker closed, with the breaker open, and with the breaker opening and closing during a sine beat test. In all cases, a successful test required the breaker to retain its status quo unless signaled to respond. Failure to maintain its status or failure to respond would be considered a failure.

To compare a floor response spectra curve to the test values, the value from the response spectra curve is divided by the Q-factor 5.5. EBASCO Services Incorporated has furnished the floor response curves at both the 19.5 elevation and the 43.0 elevation, for equipment, such as switchgear, having a 5 percent damping factor. Each of these curves shows that the peak responses occur only at periods greater than 0.2 sec. (or frequencies lower than 5 Hz) in any direction. As stated previously, the lowest natural frequency associated with the switchgear is 7 Hz and the switchgear was tested at this frequency or the greatest fundamental period. Because of the 5 percent damping factor, the maximum DBE response would then be as determined from the curves:

Direction of Motion	Elevation 19.5	Elevation 43.0
E-W	0.44g	0.39g
N-S	0.32g	0.23g
Vertical	0.36g	0.36g

The required peak input for each response is determined by dividing the "Q" value (Q = 5.5 for equipment with 5 percent damping) into the peak response specified. In the above table, the greatest input required is for the E-W motion at the 19.5 elevation, since

$$\frac{0.44g}{5.5} = 0.08g$$

The actual test input to the switchgear was 0.8g a factor of ten (10) times the requirement of 0.08g. Qualification is established on the basis that the maximum expected response as shown in the EBASCO curves is smaller than the responses actually withstood by the equipment during the tests.

4. Test Summary

1. During this series of tests, the equipment was subjected to many more earthquakes than it would ever experience during its economic life.

2. Though the equipment had been subjected to an excessive number of tests, there was no physical equipment failure.

3. At no time during the tests did the breaker trip or close, unless called upon to do so. When signaled to operate, it did so relialy every time.

4. During some of the tests, a slight bounce of a normally closed contact of an SG-relay (with the coil de-energized) was observed.

The switchgear required at this station includes 50DHP250 switchgear which has the same basic cell design as the 50DHP350 equipment, but the 50DHP250 circuit breaker has one basic difference from that of the 50DHP350 circuit breaker - namely, a smaller arc chute.

The qualification of the 50DHP250 breaker is based on the test data for the 50DHP350, comparison of physical features, and dynamic analysis. Under conditions of seismic vibration, the dynamic response in the 50DHP250 breaker will be less severe than in the 50DHP350 breaker because the pole units are identical and the effective mass of the 50DHP250 arc chute at the support is less than 80 percent of the effective mass of the 50DHP350. Thus, the 50DHP250 arc chute with its (pole unit) support is well qualified for use in any seismic environment where the 50DHP350 arc chute and its supports have been previously qualified.

Since the 50DHP350 breaker has been qualified by testing for service in the seismic environment postulated at the St. Lucie Unit No. 1 Nuclear Power Station as specified by EBASCO Services Incorporated, and since all components of the 50DHP250 breaker are as well qualified for the same seismic environment postulated as the corresponding parts for the 50DHP350 breaker, the 50DHP250 breaker is also qualified for service per the EBASCO Services specification.

5. Conclusion

The sine beat method of testing is conservative because:

1. The response from a sine beat input is more severe than from random inputs.

2. Testing is performed at all natural frequencies determined in the equipment.

3. Additional tests are performed at other selected frequencies.

From the tests and analysis presented in this report, it is concluded that the 50DHP350 metal-clad switchgear will satisfactorily withstand the maximum seismic requirements of St. Lucie Unit No. 1 Nuclear Power Station. The qualification of the 50DHP250 switchgear equipment for service at St. Lucie is established by the seismic tests performed on the 50DHP350 switchgear equipment along with analytical comparisons based upon the static loading tests and other simple measurements. The conclusion is made that the test program and the subsequent correlation and additional analysis of special features of the subject switchgear unit has verified the ability of the 50DHP250 metal-clad switchgear furnished for this station to operate satisfactorily during the specified seismic environment.

Refer to NLI report R-037088-2, revision 1 for in-cabinet response spectra which reflects the modified switchgear configuration as documented in PCM's 89350 and 89351.

B. <u>480 VOLT SWITCHGEAR AND STATION SERVICE TRANSFORMERS</u>

The following is a summary of the ITE Imperial Corporation seismic qualification of the 480 volt switchgear .

1. <u>Test Methods</u>

The 480 volt switchgear, bolted to a one inch steel plate, was mounted to the horizontal platform of the seismic shock machine. The platform was then preloaded to the desired static capacity.

Accelerometers were attached to the shock platform to monitor the horizontal force and the resultant force at 35° above the horizontal. Additional accelerometers were attached to the unit under test at the points specified and are reported in the results of test.

Once satisfactory operating conditions were established, a series of shocks were applied at various magnitudes ranging from 0.5 g's to a maximum of 3.0 g's acceleration. During each shock, the system was monitored for electrical malfunction or mechanical damage as a result of the applied acceleration. During seismic testing the circuit breakers were successfully electrically opened and electrically closed. No malfunctions were observed throughout the test (Table 3B-2A is a letter certifying this).

The above tests were performed with the shock applied in the front to back direction of the unit. The unit was then turned 90° about vertical axis and subjected to at least one shock at 3 g's in the side to side direction.

At the completion of the tests with the four-cycle system, all springs were replaced to attain the proper velocities for six-cycle shock tests and eleven-cycle shock tests.

2. <u>Results of Shock Tests</u>

The observations noted and recorded during the above detailed test procedure are listed in Table 3B-1. In 1972, additional shock tests were performed at basic frequencies of 6 and 11 cps along the front-to-back axis. The horizontal output accelerations at position number 2 were recorded on oscillographs. Refer to Table 3B-2.

3. <u>Resonant Survey</u>

At the completion of shock tests, the 480 volt switchgear was removed from the shock machine and then mounted to the table of a low frequency vibration machine.

The unit was subjected to a vibration scan in a frequency range from 5 to 33 cps. For this scan, accelerometers were attached to the unit at the same points as monitored during the shock tests. Resonant conditions of these points, if any, were noted and recorded. This scan was performed in each of the two horizontal axes.

Monitored Points	Axis	Resonant Freg. (cps)	Transmissibility [1] (output/input)
2	Front to back	19.0	29.0
3	Side to Side	12.0	3.5
4	Side to Side	10.0	1.75
5	Front to back	16.0	35.0
6	Front to back	22.0	60.0

There were no appreciable resonant conditions of Points 1 or 7, nor the structure frame, in the frequency range scanned.

In 1972, an additional vibration scan was performed along the front to back axis at position number 2 on the outside of the left front panel 10.5 inches from the top. The data is listed below:

Frequency c.p.s.	Transmissibility (ratio of output/input)
5	1.0
6	1.0
7	1.0
8	1.0
9	4.0
10	2.0
11	2.0
12	2.0
13	2.0
14	2.0
15	2.0
16	3.0
17	4.0
18	8.5
19	16.0
20	8.0
21	5.0
22	6.25
23	6.0
24	3.33
25	2.33
26	1.0
27	0.66
28	0.35
29	0.75
30	1.0
31	1.0
32	1.5
33	1.0

1) Transmissibility is defined as a ratio of the output amplitude divided by the input amplitude.

TABLE 3B-1

480 VOLT SWITCHGEAR

RESULTS OF SHOCK TEST (PEAK ACCELERATIONS)

I Four Cycle Shock Tests:

Along the Front to Back Axis:

Shock	Inputs	(g's)	Outputs		(g's)				Circuits Monitored				
No.	36°	Horiz.	1	2		Mode	EB4	EB6	EB8	EB10	EB12	EB14	EB16	EB18
1	0.6	0.5	0.5	0.5		1	ok	ok	ok	ok	ok	ok	ok	ok
4	1.0	0.8	1.1	1.1		1	ok	ok	ok	ok	ok	ok	ok	ok
7	1.9	1.6	2.4	2.2		1	ok	ok	ok	ok	ok	ok	ok	ok
10	3.0	2.5	2.6	3.2		1	ok	ok	ok	ok	ok	ok	ok	ok

Shock	Inputs	(g's)	Outputs		(g's)				Circuits Monitored					
No.	36°	Horiz.	5	6	7	Mode	EB4	EB6	EB8	EB10	EB12	EB14	EB16	EB18
12	3.0	2.5	2.4	2.7	2.6	1			Circuits not Monitored					

Along the Side to Side Axis:

Shock	Inputs	(g's)	Outputs		(g's)				Circuits Monitored					
No.	360	Horiz.	3	4		Mode	EB4	EB6	EB8	EB10	EB12	EB14	EB16	EB18
14	3.1	2.7	4.2	3.0		2	ok	ok	ok	ok	ok	ok	ok	ok

TABLE 3B-1 (Cont'd)

II Six-Cycle Shock Tests:

Along the Front to Back Axis:

Shock	Inputs	(g's)	Outputs	(g's)		Circuits Monitored							
No.	36°	Horiz.	1	2	Mode	EB4	EB6	EB8	EB10	EB12	EB14	EB16	EB18
1	0.7	0.6	1.0	0.8	1	ok	ok	ok	ok	ok	ok	ok	ok
2	1.4	1.2	1.5	1.3	1	ok	ok	ok	ok	ok	ok	ok	ok
3	2.1	1.7	2.4	2.1	1	ok	ok	ok	ok	ok	ok	ok	ok
9	3.3	2.8	4.1	4.0	1	ok	ok	ok	ok	ok	ok	ok	ok

Shock	Inputs	(g's)	Outputs	(g's)		Circuits Monitored							
No.	36°	Horiz.	6	7	Mode	EB4	EB6	EB8	EB10	EB12	EB14	EB16	EB18
10	3.2	2.8	5.1	4.4	1	ok	ok	ok	ok	ok	ok	ok	ok

Along the Side to Side Axis:

Shock	Inputs	(g's)	Outputs	(g's)		Circuits Monitored							
No.	36°	Horiz.	3	4	Mode	EB4	EB6	EB8	EB10	EB12	EB14	EB16	EB18
12	3.3	2.8	3.3	3.8	2	ok	ok	ok	ok	ok	ok	ok	ok

TABLE 3B-1 (Cont'd)

III Eleven-Cycle Shock Tests:

Along the Front to Back Axis:

Shock No.	Inputs 36°	(g's) Horiz.	Outputs 1	2	(g's)	Mode	EB4	EB6	Circuits Monitored EB8	EB10	EB12	EB14	EB16	EB18
1	0.6	0.5	0.9	0.8		3	ok	ok	ok	ok	ok	ok	ok	ok
2	1.0	0.9	1.7	1.3		3	ok	ok	ok	ok	ok	ok	ok	ok
3	1.8	1.4	3.0	2.0		3	ok	ok	ok	ok	ok	ok	ok	ok
6	2.9	2.4	4.2	4.2		3	ok	ok	ok	ok	ok	ok	ok	ok

Shock No.	Inputs 36°	(g's) Horiz.	Outputs 5	6	(g's) 7	Mode	EB4	EB6	Circuits Monitored EB8	EB10	EB12	EB14	EB16	EB18
8	3.3	2.6	4.3	4.0	6.6	3	ok	ok	ok	ok	ok	ok	ok	ok

Along the Side to Side Axis:

Shock No.	Inputs 36°	(g's) Horiz.	Outputs 3	4	(g's)	Mode	EB4	EB6	Circuits Monitored EB8	EB10	EB12	EB14	EB16	EB18
9	3.4	2.4	2.6	2.0		4	ok	ok	ok	ok	ok	ok	ok	ok
10	3.4	2.4	3.0	2.2		4	ok	ok	ok	ok	ok	ok	ok	ok

TABLE 3B-2
480 VOLT SWITCHGEAR
RESULTS OF SHOCK TEST [1] (AVERAGE ACCELERATION)

I Six Cycle Shock Tests

BLOW NO.	INPUT HORIZ. ACCEL. g	INPUT RESULT. ACCEL. g	OUTPUT HORIZ. ACCEL. g
1	0.6	0.7	0.6
2	1.1	1.4	1.1
3	1.6	2.0	1.6
4	1.9	2.2	2.0
5	2.1	2.5	2.3
6	2.3	2.7	2.4
7	2.6	3.2	2.9
8	2.7	3.2	3.0
9	2.7	3.2	2.8

II Eleven Cycle Shock Tests

BLOW NO.	INPUT HORIZ. ACCEL. g	INPUT RESULT. ACCEL. g	OUTPUT HORIZ. ACCEL. g
1	0.5	0.6	0.85
2	0.8	1.1	1.4
3	1.4	1.7	2.4
4	2.0	2.5	3.4
5	2.2	2.7	3.8
6	2.3	2.8	3.9

1 The output horizontal acceleration for blows above were interpreted from oscillograph film an the basis of average acceleration and differ from values listed on Table 3B-1, which were peak values.

4. Mounted Equipment

Floor response spectra curves provided for usage on this equipment give a maximum DBE acceleration of 0.25g input to devices mounted on the equipment for 19 Hz resonance. The following devices require evaluation for this acceleration input.

B/M Pc. No.	Device	Withstandability	Remarks
50/51	IAV	> 0.25g	Certified
59	EOB Crkt. Bkr.	> 0.25g	Certified
33	C77	> 0.25g	Certified

5. Station Service Transformer

The following is a summary of the BBC Brown Boveri seismic qualification of the 480 volt station service dry transformer.

The transformer was mounted to a triaxial seismic test table. The test commenced with a series of three sine wave resonance exploratory scans, one in each orthoganal axis. The scanning rate was one octave per minute between 1 and 50 Hz at 0.2g. acceleration. The next seismic test was the first of five operating basis earthquake tests. For the seismic tests the transformer secondary was energized so that primary voltage appeared at the primary terminals. This test enveloped the Required Response Spectrum. However, it was noticed that the low and high voltage air terminal chambers had a tendency to flap up and down on the table since they had not been separately welded down. At this point they were welded by one inch welds between the lifting eyes on the bases of the air terminal chambers. It was also noticed that the displacement measuring linear variable differential transformer mounting was in motion. The mounting was strengthened.

The next four operating basis earthquake tests were performed without event. It was observed, however, that the differential transformer mounting continued to be in motion, invalidating the displacement measurements taken.

Following the five operating basis earthquake tests a design basis earthquake test was performed. No anomalies were noted.

Subsequent to the required tests reported above an optional test at the discretion of BBC Brown Boveri (the manufacturer) was performed at table limits in the region of the maximum of the required response. Levels above 10g.'s were recorded for the table motion. No damage was sustained by the transformer, the mountings, or the transformer enclosures.

At the conclusion of the qualification seismic tests, the transformer was returned to the Bland, VA transformer manufacturing plant where it was retested for all performance requirements. All the readings were within the limits of the ANSI requirements for dry type transformers.

ITE Imperial
CORPORATION

May 23, 1974

Ebasco Services, Incorporated
Two Rector Street
New York, New York 10006

Attention: Mr. L. J. Mulligan

Subject: Seismic Certification for Switchgear
Florida Power & Light Company
Purchase Order NY-422246
I-T-E Shop Order 33-47095

Gentlemen:

This is to certify that I-T-E Type K Low Voltage Circuit Breakers have been seismic tested and are certified for use at the Florida Power & Light, St. Lucie Station. During seismic tests the circuit breakers were successfully electrically opened and electrically closed.

I-T-E IMPERIAL CORPORATION

Thomas E. Feil

Thomas E. Feil

TEF/cr

C. SEISMIC QUALIFICATION OF 480 VOLT MOTOR CONTROL CENTERS

The following is a summary, of General Electric's Report No. 70ICS100, dated September 3,1970, which qualifies the 480 volt motor control centers used on St. Lucie Unit 1.

1. Abstract

These tests indicate that the 7700 line motor control center is suitable up to at least 0.5g base input accelerations through a frequency band width from 5 to 500 Hz.

2. Description of Tests

The test article was attached to the vibration table using conventional bolting methods typical of actual installation techniques. The vibration fixture (table) was non-resonant within the frequency band width of interest.

Accelerometers monitored input acceleration at the base of the equipment and resulting response accelerations at significant points within the test article. Each response monitor point was instrumented to detect vibration acceleration response in the direction of the input forcing (base) vibration.

The test article was swept in frequency from 5 to 500 Hz at a one-half octave per minute sweep rate, at a constant 0.5g input acceleration.

The equipment was vibrated in each of its three orthogonal axes; vertical, horizontal, in-breadth, and horizontal fore-and-aft.

All vibration sweeps were made with a 480 volt ac 60 Hz source connected to the control center main bus.

Vibration sweeps were made in two modes of functional status:

a) with each starter unit disconnect in the ON position, but starter not energized.

b) with each starter energized.

The equipment was vibrated with all doors removed from the enclosure to facilitate observation of component behavior during tests. The doors add little to the rigidity of the structures, and their removal did not significantly alter test results.

The input acceleration levels were held at a constant 0.1 inch double amplitude displacement from 5 Hz to 10 Hz instead of a constant 0.5g level due to machine limitations. The date was normalized and this deviation does not affect the accuracy of the data acquired.

3. Test Results and Discussion

Plots of input and response acceleration levels for the three orthogonal axes of vibration were made for various accelerometer locations. The plots reflect data recorded during the de-energized mode of starters, and represent worst case conditions insofar as response acceleration levels are concerned.

No changes in functional status were observed during the vibration sweeps: i.e., starters in the de-energized mode did not close, and starters in the energized mode did not oven.

An examination of the plots shows a prominent resonant point, at 5-6 Hz, when the equipment was vibrated in both horizontal directions. This resonance at all monitored points represents the structure resonant frequency. Resonances at various higher frequencies represent individual component resonant points.

Table 3B-3 summarizes maximum acceleration levels between 5 and 100 Hz measured at each response point, as taken from plots. An examination of the data of Table 3B-3 indicates the following:

a) highest response accelerations were generallv found, as would be expected, at brackets attached to starter unit frames (Pushbutton brackets, cantilever mounted, are particular examines) .

b) no discernible pattern is evident to relate the vertical position of a starter unit within the structure to the response acceleration of the unit.

c) disregarding the relatively high acceleration responses of pushbutton brackets, all monitored points display reasonably similar maximum response levels in any given direction of input acceleration (maximum recorded difference indicates an approximate 3:1 ratio of response level).

4. Conclusions

a) The tests conducted indicate the equipment as tested is suitable for applications up to at least 0.5g base input accelerations through a frequency band width from 5 to 500 Hz.

b) The resonant level of the equipment structure appears to be at 5-6 Hz, while component resonant levels are at higher frequencies.

c) The devices can be satisfactory operated when subjected to the seismic testing described above. (Table 3B-2B is a letter certifying this).

GENERAL ELECTRIC COMPANY

INDUSTRY CONTROL DEPARTMENT

1501 ROANOKE BOULEVARD, SALEM, VIRGINIA . . . . TELEPHONE 389-7011

Certification of Compliance

For

GE Company Requisition 300-91088

Ebasco Services, Inc.

For: Florida Power & Light Company

Hutchinson Island, Florida

This is to certify that the Class 1 Motor Control Centers furnished on this requisition meet the seismic requirements of Ebasco Specification 210-69 as revised July 20, 1971. During testing as described in Technical Report 70IC8100, the

Thomas J. ______ Mgr., Industrial Group Control Engineering

Sworn to and subscribed before me this 23 day of May 1974.

Brenda ______
Notary Public

My Commission Expires June 25, 1974

TABLE 3B-2B (Cont'd)

NOTE ON SPECIFICATION 210-69

Ebasco specification 210-69 (July 20, 1971) specified the following:

"If a seismic design is specified in Part One, Seller shall provide test data to demonstrate the adequacy of his product to withstand the effects of the specified seismic forces. A product which complies with the intent of this requirement shall after exposure to specified seismic forces:

a) Exhibit no undue deflection which would prevent any component specified in this specification from performing normal uninterrupted operation.

b) Have no components dislocated, which would prevent uninterrupted normal operation (i.e. fuse thrown out of fuse holder, bolt used to mount control transformer sheared, etc.).

c) Maintain all components in the same operating position during the disturbance as they were prior to it.

d) Permit operation of all components during the disturbance, i.e. if starter is energized or deenergized during the disturbance, it will react accordingly.

In testing, the horizontal and vertical accelerations shall be applied simultaneously. The application of the combination of these two forces shall be repeated to simulate horizontal acceleration of the gear, in as many directions as necessary to demonstrate the adequacy of the gear due to its asymmetry. Response spectra curves for the ground motion due to earthquake are attached and form part of this specification."

TABLE 3B-3

MOTOR CONTROL CENTERS
MAXIMUM RESPONSE ACCELERATIONS TO 0.5g BASE INPUT ACCELERATION
(BETWEEN 5 & 100 Hz)

ACCEL. LOCATION	VERTICAL (g)	HORIZONTAL IN-BREADTH (g)	HORIZONTAL FORE-AND-AFT (g)
SZ. 4 FVNR UNIT FRAME	1.7	1.2	2.0
SZ. 4 RVNR UNIT	1.3	.95	1.3
SZ. 4 RVNR STARTER BASE	1.4	.80	1.5
SZ. 4 RVNR PUSHBUTTON BRACKET	5.5	.80	1.7
SZ. 3 FVNR DISCONNECT BRACKET	2.2	.90	.75
SZ. 3 FVNR UNIT FRAME	2.1	.90	.60
SZ. 1 FVNR DISCONNECT BRACKET	2.4	.85	.85
SZ. 3 FVNR PUSHBUTTON BRACKET	8.4	1.4	8.1
SZ. 4 RVNR UNIT FRAME	1.5	.72	1.1
SZ. 4 RVNR DISCONNECT BRACKET	2.8	.85	.85
SZ. 2 FVNR UNIT FRAME	1.9	.95	1.1

D. SEISMIC QUALIFICATION OF 125 VOLT AC AND DC PANELS

I. AC PANELS

The following is a summary of the ITE Imperial Corporation Test-R-STD-5, "Seismic Certification CDP-4 Distribution Panel Boards", dated November 2, 1972. The test seismically qualifies power panels: PP-101, 102, 103, 110, 111, 112, and 114; maintenance bypass buses 1A and 1B, and instrument buses 1MA, 1MB, 1MC and IMD.

1. Input Accelerations

The maximum input acceleration determined as the high frequency asymptotic level is less than 0.20 g, i.e., zero period value. A horizontal input of 0.3 g and a vertical input of 0.1 g is required. Therefore, in order to ensure that all the requirements and the acceleration levels are met, a minimum horizontal input acceleration level of 0.3 g is used as a conservative input value for a continuous sinusoidal input function.

2. Discussion and Test Results

The test results are based upon a continuous sinusoidal motion of long duration. The specimen was tested simultaneously in the horizontal and vertical axis. The ratio of the horizontal to the vertical component was 1 to 0.726 or the tangent of 36 degrees. All acceleration values are the horizontal input measured directly at the drive point of the test specimen. Therefore, any response of the testing system was eliminated and only the actual input acceleration to the specimen was measured. The acceleration values are input g's and are not to be confused with response or output g's.

The contacts were monitored during all tests. The specimen was tested with both open mode and closed mode conditions.

Sample components/devices have been tested separately as a bare device as well as tested in their structures.

The frequency range investigated was between 1 Hz through 33 Hz. For the components, no resonance points were determined in the range.

No point of resonance was determined on the structure in the seismic critical range (1 to 20 Hz). The specimen, therefore, would experience very little amplification and see basically the static loading from the seismic event. The specimen, however, was dynamic tested, with a continuous sinusoidal and random vibration input of g forces greater than the required values (i.e., received a greater loading than would be experienced by the seismic event).

During the main withstandabilitv tests (dwell tests), horizontal acceleration inputs of greater than 1.0 g was reached, i.e., vertical input was greater than 0.73 g's.

During the supplementary random vibration tests, horizontal input acceleration of greater than 2.5 g's was reached. In this test function the center band frequency was set at 4 Hz with a band width of $\pm$3.16 Hz, i.e., 0.84 Hz to 7.16 Hz. The duration of random vibration test was sixty (60) seconds.

3. Summary

Sample equipment representing the subject equipment has been thoroughly investigated for its seismic withstandability. Seismic tests were conducted on the sample equipment with limits well in excess of the referenced requirements.

At no time during any of the tests conducted was there any indication of malfunction or failure. The subject equipment meets well in excess the referenced requirements.

II. DC PANELS

The following is a summary of the ITE Imperial Corporation Test-R-STD-2E, "Seismic Certification-Free Standing FC-20 Switchboards", dated November 2, 1973.

1. Input Accelerations

The maximum input acceleration being determined as the high frequency asymptotic level is less than 0.20 g, i.e., zero period value. A horizontal input of 0.3 g and a vertical input of 0.1 g is required. Therefore, in order to insure that all the requirements and the acceleration levels are met, a minimum horizontal input acceleration level of 0.3 g is used as a conservative input value for a continuous sinusoidal input function.

2. Discussion and Test Data

The test results are based upon a continuous sinusoidal motion of long duration. The specimen was tested simultaneously in the horizontal and vertical axis. The ratio of the horizontal to the vertical component was 1 to 0.726 or the tangent of 36 degrees. All acceleration values are the horizontal input measured directly at the drive point of the test specimen. Therefore, any response of the testing system was eliminated and only the actual input acceleration to the specimen was measured. The acceleration values are input g's and are not to be confused with response or output g's.

The contacts were monitored during all tests. The specimen was tested with both open mode and closed Mode conditions.

Sample components/devices have been tested separately as a bare device as well as tested in their structures.

The frequency range investigated was between 1 Hz through 30 Hz. For the components, no resonance points were determined in the range.

The resonance frequency of the structure was determined to fall in the range of 8.7 Hz to 9.9 Hz with a minimum damping factor of 7.7 per cent at 0.1 g input and maximum damping factor of 11.8 per cent at 0.1 g input - depending on axis of excitation. The specimen, however, was dynamic tested, at the resonant region and other frequencies, with a continuous sinusoidal and random vibration input of g forces greater than the required values (i.e., received a greater loading than would be experienced by the seismic event).

During the main withstandability tests (dwell tests), horizontal acceleration inputs of greater than 1.0 g was reached, i.e., vertical input was greater than 0.73 g's.

During the supplementary random vibration tests, horizontal input acceleration of greater than 1.0 g was reached. In this test function the center band frequency was set at 8 Hz and 10 Hz with a band width of ±3 Hz, i.e., 5 Hz to 11 Hz, and 7 Hz to 13 Hz respectively. The duration of random vibration test was sixty (60) seconds.

3. Summary

Sample equipment representing the subject equipment has been thoroughly investigated for its seismic withstandability. Seismic tests were conducted on the sample equipment with limits well in excess of the referenced requirements.

At no time during any of the tests conducted was there any indication of malfunction or failure. The subject equipment meets well in excess the referenced requirements.

E. SEISMIC QUALIFICATION OF ESF ACTUATION AND MEASUREMENT CABINETS

The original ESFAS panels have been vibration tested. Circuits were energized during the test; circuits were actuated before and after testing; and circuits did not change state during the test. The ESFAS relays were vibration tested. No natural frequencies were encountered between 5 and 33 Hz; no natural frequencies exist below 5 Hz; and the relay performed properly and met or exceeded all specified requirements from 5 to 33 Hz. With regard to humidity considerations, the relays are evacuated and are hermetically sealed. Thus, the effects of variations in humidity have been accommodated by design. (Similar relays have been qualified for humidity - see certification, pages 3B-30, 31, and 32) A summary of the vibration tests is provided below for both the ESFAS panels and relays. The vendor's certification for the ESFAS equipment is attached.

The Original ESFAS Panels - Test Summary[1]

The original ESFAS cabinet design utilizes S. H. Couch relays model number 4CP36-AF. These relays were selected because of their inherent capability to accommodate vibratory motion without loss of required function. They are of the same basic design as relays utilized by the U. S. Navy. The relays furnished the navy were successfully tested to accommodate vibratory motion in 1969 (see Consolidated Controls Corporation letter of February 14, 1974 provided with this appendix). Since there were minor differences in design the 4CP36-AF relays were installed in the ESFAS cabinets and the total cabinet assemblies were vibration tested as summarized below and detailed in reference (1). Relays were energized during these tests and they maintained required positions during the test period.

a) Procedure

1) The 9N17 cabinets will be bolted together and fixtured so as to simulate normal operating attitude and will be secured to the platform of the seismic test machine.

2) The above will be powered from a 115VAC, 60 Hz source with all input energized. The channel values for this test will be as follows:

Containment press	-	14.7 psia
Containment radiation	-	100 mr/hr
Refuel level	-	30 feet
Pressurizer pressure	-	2130 psig
SGIA pressure	-	800 psig
SGIB pressure	-	800 psig

3) The system bistables will be set as follows:

BA101 - 30 psia	BA105 - 15 feet
BA102 - 35 psia	BA106 - 1800 psig
BA103 - 20 psia	BA107 - 1900 psig

BA104 - 500 mr/hr	BA108 - 400 psig
BA109 - 600 psig	BA111 - 600 psig
BA110 - 400 psig	

4) Logic inputs to the actuation cabinet will be supplied by jumpering the channel 1 logic to channel 2 inputs and the channel 3 logic to channel 4 inputs.

5) Incident monitoring will be provided for:

SIAS	SGIA
RAS	CSAS
CIS	SGIB

The test specimen will then undergo the following test procedure:

(a) The test specimen will be subjected to a resonance survey in each of three mutually perpendicular planes.

(b) Plane I - Lateral for a frequency range of 2 to 20 Hz at an acceleration amplitude of 0.8 g's. Foundation amplitude varies from 4 inches at 2 Hz to 0.04 inches at 10 Hz. The cabinets will be visually and audibly observed for resonances during this sweep test. All resonant frequencies will be noted and recorded.

(c) Plane II - Longitudinal for a frequency range of 2 to 20 Hz at an acceleration amplitude of 0.8 g's. Foundation amplitude will vary from 4 inches at 2 Hz to 0.04 inches at 20 Hz. The cabinets will be audibly and visually observed for resonances during this sweep test. All resonant frequencies will be noted and recorded.

(d) Plane III - Vertical for a frequency range of 2 to 20 Hz at an amplitude of 0.6 g's. The foundation amplitude will vary from 3 inches at 2 Hz to .03 inches at 20 Hz. The cabinets will be visually and audibly observed for resonant frequencies during this sweep test. All resonant frequencies will be noted and recorded.

(e) Electrical criteria for this test will be no bistable trips of logic change of state for SIAS, CIS, CSAS, RAS, SGIA, SGIB.

b) Conclusions and Recommendations

With the exception of the relay panels becoming loose and opening, the 9N17-1, -3, -5, assembly performed to the requirements of the applicable specifications. The entire assembly was energized before, during and after the test period, and at no time was a functional discrepancy noted.

Upon review of all pertinent data, CCC considers the 9N17 system qualified to the Class I criteria when the following recommendation is completed:

Recommendation No. 1:

All relay front panels are to be secured to the cabinet frame with at least two additional screws positioned near the top of the panel, on each side. These screws are in addition to the present fasteners at the top of the panels.

This recommendation has been implemented.

ESFAS Relays - Test Summary[2]

The exact relays installed in the ESFAS were tested separately as described below and in reference (2). During the test program the relay was signaled to change state (during the period of applied vibratory motion) at each of the frequencies specified below. At no time during the tests did the relay fail to close or open when called upon to do so, nor did it close or open when not called upon to do so. In summary, the exact relay was tested to insure its ability to accommodate vibratory motion and, as expected, it performed satisfactorily.

a) Equipment Identification

S.H. Couch Relay	4CP36-AF SN/7250
Socket	29-407639-01
Spacer (2)	R1090

b) Equipment Specification

1) EBASCO Services FLO-8770.145

   Engineered Safeguard Logic Panels.

2) Requirements per Paragraph 4.3 of Specification

   Horizontal 0.8 G 2-20 Hz.

   Vertical 0.6 G 2-20 Hz.

3) Equipment must operate during and after seismic event (vibration).

c) Test Method

1) Mount the specimen relay and chassis on the vibration test machine in Plane I. Photograph the test setup.

2) Attach the accelerometer to the test chassis in close proximity to the base of the relay or on the relay itself.

3) Connect a 24v DC power supply, switching device and strip chart recorder in such a manner to energize the relay approximately once each 5 seconds. Monitor both the coil voltage and a normally open set of contacts to show operation of the relay during vibration.

4) Sweep Test

Run a sweep test from 5 to 33 Hz per paragraph C.5 to check for any natural frequencies. If no natural frequencies are found between 5 and 33 Hz, it has been determined that there are no natural frequencies 5 Hz or below 5 Hz for this device.

5) Continuous Test

(a) Apply a continuous sinusoidal vibration to the sample from 5 Hz to 33 Hz as follows:

Frequency Hz	Acceleration G	Displacement (D.A.) inches
5	0.82	.65
6	1.2	.65
7	1.7	.65
8	2.5	.75
9	3.2	.75
10	4.0	.75
11	4.9	.75
12	5.8	.75
13	6.5	.75
14	7.9	.75
15	9.0	.75
16	10.0	.75
17	10.0	as required
18	10.0	as required
19	10.0	as required
20	10.0	as required
21	10.0	as required
22	10.0	as required
23	10.0	as required
24	10.0	as required
25	10.0	as required
26	10.0	as required
27	10.0	as required
28	10.0	as required
29	10.0	as required
30	10.0	as required
31	10.0	as required
32	10.0	as required
33	10.0	as required

(b) Apply vibration for a minimum of 30 seconds at each frequency. Operate relay with switching mechanism and record relay contact and coil results on the strip chart recorder.

d) Results and Conclusions

1) The relay performed properly and met or exceeded all specified requirements from 5 to 33 Hz.

2) No natural frequencies were encountered between 5 and 33 Hz. It has been determined that no natural frequencies exist below 5 Hz based upon the relay configuration and test data.

ESFAS Relays - Cabinet Amplification Summary [3]

The NRC in its supplement to the SER of May 1975 at Section 8.3.3 questioned the validity of the tests performed in reference 2 with respect to the possible amplification of input g values for relays mounted in the ESFAS cabinets.

The relay panel was analyzed (reference 3) to determine its natural frequency. The calculated fundamental frequency of the panel was found to be 16.7 Hz. The cabinet with the relays was also subjected to a shake table test (reference 1). The test results indicated that the overall cabinet framing natural frequency was 20Hz or over. The cabinet relay resonance frequency of 20 Hz compares favorably with the calculated value of 16.7 Hz.

The floor response spectra for elevation 6.2 of the reactor auxiliary building are provided by Figures 3.7-21 and 23. From these figures it is evident that components with natural frequencies greater than 6.7 Hz will not experience resonances. Therefore, the cabinet and all of its components will respond as a rigid body to a maximum floor acceleration of about 0.2g, i.e., the cabinet will not amplify the floor response. In view of the 10 g accelerations used in the tests of reference 2, it is clear that the relays have been successfully qualified at seismic levels well above the design levels for St. Lucie 1.

<u>References</u>

1) Consolidated Controls Corporation, Engineering Report No. 824, "Seismic Test Report for Engineered Safeguards Panels - St. Lucie Nuclear Power Station Unit #1," December 5, 1973.

2) Consolidated Control Corporation Engineering Report No. 862, "Seismic Test Report of 4CP36-AF Relay Mfg. by S. H. Couch Div - ESB Safety Features Actuation System" May 7, 1974.

3) Consolidated Control Corporation, Engineering Report No. 912, "Second Addendum to Seismic Test Report for Engineered Safeguards Panels - St. Lucie Nuclear Power Station Unit #1," May 23, 1975.

CONSOLIDATED CONTROLS CORPORATION

15 DURANT AVENUE, BETHEL, CONNECTICUT 06801 203 743-6721

TWX 710-456-0446 TELEX 967615

A SUBSIDIARY OF CONDEC CORPORATION

February 14, 1974

Ebasco Services Inc.
2 Rector Street Room 800
New York, New York 10006

Attention: Mr. Vito Oniunas

Subject: Qualification of Engineered Safeguard

Gentlemen:

Enclosed is the certification of the humidity and seismic qualification items per our telecon of 2/12/74.

We are pleased to be of assistance at this time.

Very truly yours,

CONSOLIDATED CONTROLS CORPORATION

G. L. Schoenbaum
Product Manager, Systems

/lmp

Enclosure: Certification
Engineered Safeguard Logic Panels
St. Lucie Nuclear Power Station
Unit # 1
Dated February 14, 1974

CERTIFICATION
Engineered Safeguard Logic Panels
St. Lucie Nuclear Power Station
Unit #1
Februrary 14, 1974

I Engineered Safeguard Logic modules 6N81 through 6N87 similar to 6N88, 6N89, 6N90, 6N91 and 6N92 were temperature qualified to 130 degrees Fahrenheit and 94% RH. The modules met all specification during the seven day test. Consolidated Controls Corporation Engineering Report # 803 dated 2/21/73 incorporates the procedure and data of this qualification.

II Engineered Safeguard Logic actuation relays S. H. Couch type 4CP36-AF were tested both in the energized and de-energized state during seismic qualification of the St. Lucie I equipment. Proper operations of the relays were verified prior to and subsequent to the application of the seismic forces. This data is contained in Consolidated Controls Corporation Engineering Report # 824 dated 12/5/73.

In addition to the above qualification testing, Consolidated Controls Corporation has previously qualified actuation relays built by S. H. Couch Company. Consolidated Controls Corporation Engineering Report #771, Confidential Restricted Data dated 12/5/69, documents the operation of S. H. Couch 4AP37-AF relay during a MIL-STD-167 vibration test. After three hours of vibration, two relays were switched due to the application of trip imput driving signals. The equipment tested was Reactor-Protective Equipment being provided to the U. S. Navy. The 4AP37-AF relay is of the same construction as the 4CP36-AF relay being used in the St. Lucie I equipment. The only differences are the "A-P" which changes contact current capacity and the "37-36" which changes the coil impedence. The relay operated properly during the vibration and met all specified performance requirements.

CONSOLIDATED CONTROLS CORPORATION

G. L. Schoenbaum
Product Manager, Systems

/lmp

F. SEISMIC QUALIFICATION OF BATTERY CHARGER

The following is a summary of tests performed by TII Testing Laboratories entitled "Report of Seismic Shock Test on One (1) Battery Charger for C & D Batteries Division of Electra Corporation - Plymouth Meeting, Pennsylvania," dated September 29, 1970.

1. Vibration Survey

Prior to the seismic shock tests, the battery charger was subjected to a vibration scan in each of the three mutually perpendicular axes, in a frequency range from 5 to 55 cps. The following resonant and/or natural frequencies were noted and recorded:

Axis	Resonant Frequency	Transmissibility
Vertical	No appreciable resonance noted	
Front to Back	27 cps	3.75 - 1
Side to Side	27 cps	5 - 1

2. Test Procedure

One battery charger, bolted to a one-inch steel plate, was mounted to the table of the Seismic Shock Machine.

Accelerometers were attached to the shock machine platform to monitor the vertical and horizontal accelerations. One accelerometer was attached to the unit under test at the top front of the charger.

Once satisfactory operating conditions were established, the unit was subjected to a series of seismic shocks applied through the front to back direction of the unit at an angle of 33 ± 2° from the horizontal. The magnitude of shock acceleration was increased to a maximum of 1.96 gravity units horizontal simultaneously and linearly combined with 1.28 gravity units vertical. After each blow, the unit was visually examined for evidence of physical or operational damage.

At the completion of this portion of the test, the battery charger, with base plate, was reoriented 90° about its vertical axis and again secured to the shock machine. The charger was subjected to at least one seismic shock in the side to side direction at the maximum acceleration loadings as described above.

The fundamental frequency of the seismic shock was 10 cps.

3. Results of Seismic Shock Tests

The battery charger was energized prior to and after shock along the front to back direction. There was no apparent physical or operational damage.

The following accelerations were recorded during these shocks:

Shock No.	Input Horizontal (g's)	Input Vertical (g's)	Output at Top of Battery Charger (g's)
1	0.5	0.4	1.1
2	0.9	0.9	1.2
3	1.2	1.0	2.0
4	1.5	1.4	1.8
5	1.7	1.5	2.5
6	2.0	1.6	2.6

The battery charger was energized at no load during the tests in side to side direction. There was no apparent physical or operational damage to the unit.

The following accelerations were recorded during these shocks:

Shock No.	Input Horizontal (g's)	Input Vertical (g's)	Output at Top of Battery Charger (g's)
1	0.6	0.4	0.9
2	0.9	0.7	1.4
3	1.2	1.0	1.6
4	1.9	1.6	2.2
5	2.0	1.7	2.4

4. Conclusions

There was no apparent physical or operational damage to the battery charger as a result of simultaneous accelerations of at least 1.96 g's horizontally combined with 1.28 g's vertically during the shock tests at a fundamental frequency of 10 cps.

G. Seismic Qualification of Station Auxiliary Motors

Pump motors have been reviewed with regard to seismic forces. The vendor's certification that the category 1E station auxiliary motors as well as appurtenances will withstand the seismic forces specified in the component specifications is provided as Table 3B-4.

TABLE 3B-4

GENERAL ELECTRIC

POWER TRANSMISSION AND DISTRIBUTION SALES DIVISION

GENERAL ELECTRIC COMPANY . . . . . . 641 LEXINGTON AVENUE
NEW YORK, NEW YORK 10022, Phone (212) 750-2617

May 24, 1974

Ebasco Services, Inc.
Two Rector Street
New York, NY 10006

Attention: Mr. L. J. Mulligan

Subject: EBASCO SERVICES, INC.
FLORIDA POWER & LIGHT
ST. LUCIE NO. 1
STATION AUXILIARY MOTORS
ITEMS 2 & 5
ORDER NO. P.O. NY-422208 - GE REQN. 300-94977

Gentlemen:

This is to certify that the subject motors will withstand the specified seismic forces as covered in your specifications. Our original analysis did not specify that the space heaters, conduit boxes and other appurtenances of the motors coverd by Items 2 and 5 would meet these conditions. We have accertained that these items meet these specifications in their entirety.

Very truly yours,

W. G. Gundersen

N. G. Gundersen
Power Transmission Sales

NGG:gl

APPENDIX 3C

ANALYSIS OF STEAM AND FEEDWATER LINE BREAK OUTSIDE CONTAINMENT FOR ST. LUCIE UNIT 1

INTRODUCTION

In analyzing a main steam or feedwater line break outside containment, particular attention is expended in determining the effects on safety related equipment.

The main steam and feedwater lines for each St. Lucie Unit are routed from the containment building to the turbine building via two seismic Class I trestles (each trestle supports a main steam line and its corresponding feedwater line). Once outside the containment building, there is no other enclosure through which the lines pass on route to the turbine building.

The main steam lines are separated by approximately 15 ft as they emerge from the containment and diverge such that at the main steam line isolation valves the lines are approximately 40 ft apart.

The only other safety related components in the area are the three auxiliary feedwater pumps and motors which are located under the trestles. The two motor driven auxiliary feedwater pumps are located under one trestle and the steam turbine driven pump is located under the other trestle.

ANALYSIS

The following analysis is based on the AEC issued "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment".

1. Protection against pipe whip has been provided outside containment for the main steam and main feedwater systems based on the following criteria:

    a. Maximum operating pressure and temperature for the main steam (MS) system is 885 psig and 52OF respectively; maximum operating pressure and temperature for the main feedwater (FW) system is 1050 psig and 44OF respectively.

    b. The auxiliary feedwater system pumps (1 steam turbine driven pump, 100 percent capacity, and 2 electric motor driven pumps, each 50 percent capacity) are located under the seismic Class I trestles that support the MS and FW piping on route from the containment to the turbine building.

---

* Pursuant to the requirements of IE Bulletin 79-01B a re-evaluation of the environmental qualification of electrical equipment installed in the plant was performed. This updates the information provided in this appendix. See Section 3.11 for referencing to FPL responses to the bulletin.

  Information provided in this appendix is historical and shall not be updated; however, it may still be similar to the re-evaluation documentation if no changes have taken place.

c. If unrestrained, a rupture in the MS or FW piping could cause a pipe whip into safety related structures and equipment (containment structure, auxiliary feedwater system components).

d. The energy level of a MS or FW whipping pipe can be demonstrated to be sufficient to impair the safety function of the auxiliary feedwater system.

2. The design criteria employed throughout the plant for the design of restraints and their spacing are predicated on restraining the individual pipe regardless of the location or orientation of all postulated ruptures. To achieve this, tables and charts were developed which indicate the maximum spans allowable between restraints to prevent development of a plastic hinge and concomitant pipe whip. The tables and charts were prepared based on calculated load combinations which can be expected to result from either circumferential or longitudinal ruptures occurring along straight runs of piping, after elbows, etc. The tables and charts take into account variables such as size and configuration of the lines and the contained energy.

3. Addressed in section 2 above.

4. A static analysis which combined the design loads of the piping at operating conditions with the loads calculated to exist after a pipe rupture was utilized in the design of the pipe whip restraints and their spacing. The rupture loads are given below:

	Reaction Force Circumferential Break	Reaction Force Longitudinal Break
Main Steam Line	697,000 lbs	435,000 lbs
Feedwater Line	43,300 lbs	26,300 lbs

Both the circumferential and the longitudinal breaks considered above are based on a flow area equivalent to the cross-sectional area of the pipe. Refer to Section 3.6 of the Safety Analysis Report for a complete description of pipe whip analysis method.

5. The main steam and feedwater lines for steam generators 1A and 1B are run on separate seismic Class I trestles. The two steam lines are separated by a distance of approximately 15 ft 7 inches center pipe to center pipe as they emerge from the containment and by a distance of 40 ft at the riser where they turn to enter the turbine building. The feedwater lines are run at a distance of 15' 7" from their respective steam lines and are located approximately 60 ft from each other at the entrance to the containment. Refer to pages 3C-11 & 3C-12.

The main steam and feedwater lines are anchored approximately 9 ft from the containment wall. In addition, pipe whip restraints have been provided on the steam lines at intervals of 15' 2", 29' 7" and 41' 9" from the anchor. The feedwater line restraints have been provided at intervals of 13' 0", 26' 0" and 38' 3" from its anchor. The spacing of these restraints on the steam and feedwater lines prevents the development of a plastic hinge at any of the supports.

All of the above restraints are designed to restrict motion normal to the axes of the MS and FW lines. The lines, therefore, will not whip against each other and in no case will the impingement pressure generated by a rupture of a line on trestle A affect the lines on trestle B. For instance, the maximum impingement pressure which steam line A could impart to its counterpart on trestle B is 14 psi. The resulting stresses are within design values for a faulted condition.

The entire seismic Class I portion of the MS and FW lines outside containment are located on the trestles. One additional restraint has been provided on each of the MS and EW lines beyond the seismic Class I portions of the pipe to prevent a rupture along the non-seismic Class I portion of pipe from adversely affecting the safety related parts.

The only safety related equipment that could be affected by a rupture in the MS or FW lines are the three auxiliary feedwater pumps (2 motor driven, 1 steam driven) which are located under the trestles. The two electric motor driven pumps are located approximately 15 ft from each other under one trestle and the steam turbine driven pump is located under the other trestle. The orientation of the MS and FW lines and the auxiliary feedwater pumps can be seen on pages 3C-11 & 3C-12 (for detail see drawing 8770-G-149, sheet 2).

Each of the two motor driven pumps supplies feedwater to one steam generator and together are capable of providing sufficient quantities of water for reactor cooldown to 300F*. The pumps take suction from the condensate storage tank and are powered from the emergency diesel generator sets in case of a loss of normal power. The turbine driven pump is capable of supplying auxiliary feedwater to both steam generators and its capacity equals that of both motor driven pumps. The turbine driven pump also takes suction from the condensate storage tank and steam power for the turbine is supplied from either main steam line upstream of the isolation valve.

Assuming a slot break on the underside of either MS or FW line, it can be shown that the maximum jet impingement pressure which could be imparted to an auxiliary feedwater pump directly below

* Note that shutdown cooling entry temperature has been raised to 325°F T_{AVE} per Technical Specification Amendment #28.

it is 55 psi and 7 psi respectively. These resultant stresses are of insufficient magnitude to have any deleterious effects on the pump.

Assuming a rupture in a main steam line and assuming an adiabatic expansion of the escaping steam, the temperature of the steam will decrease to approximately 320F upon release from the steam line. This situation can be assumed to exist for a total of from 60 to 95 seconds (depending on initial power level) during which time the effected steam generator blows dry. (We assume a loss of normal feedwater since this is the only condition which would require the use of the auxiliary feedwater system). The maximum temperatures will only be experienced by the pump towards which the jet is directed.

The equipment manufacturers for the pumps and pump motors have stated that their equipment can function in the ensuing environment described above with the only possible ill effects being the failure of pump seals due to the temperature. This type of failure could result in the loss of a maximum of 5 to 10 gpm but no loss of function.

There is also no danger that a rupture of a steam line or feedwater line could cause a loss of function of more than one auxiliary feedwater pump due to flooding. Each of the three pumps are provided with a flood wall around them to elevation +22 ft. with an access opening that would preclude any water buildup. Under normal conditions accumulation within the enclosure is impossible since the condensed steam will run out over plant grade.

There is no other credible postulation of interaction between a ruptured main steam line and any connected branch line that could lead to a more detrimental condition than that described above or otherwise affect the plant capability for safe shutdown.

6. Stresses were analyzed in the steam trestle using the working stress method. The load combinations and allowable stresses are as follows:

Loading	Stress
Dead Load + Live Load + Thermal Load + Seismic Load (Operating Basis Earthquake)	- Allowable stress per AISC Code (A-36 steel)
Dead Load + Pipe Break Load	- 1.5 x allowable stress per AISC code (A-36 steel)
Dead Load + Thermal Load + Seismic Load (Design Basis Earthquake)	- 1.5 x allowable stress per AISC Code (A-36 steel)

The seismic load factors applied for dead and equipment load are as follows:

elevation 36' 0" =	0.125	vertical
	0.25	horizontal
elevation 62' 0" =	0.25	vertical
	0.50	horizontal

7. Loadings on the restraints and hence the trestles are shown in Tables 1 and 2 on the following pages.

8. Other than the containment structure and the main steam trestles, there are no other seismic Class I structures which can credibly be affected by a rupture in the MS or FW lines.

9. Not applicable.

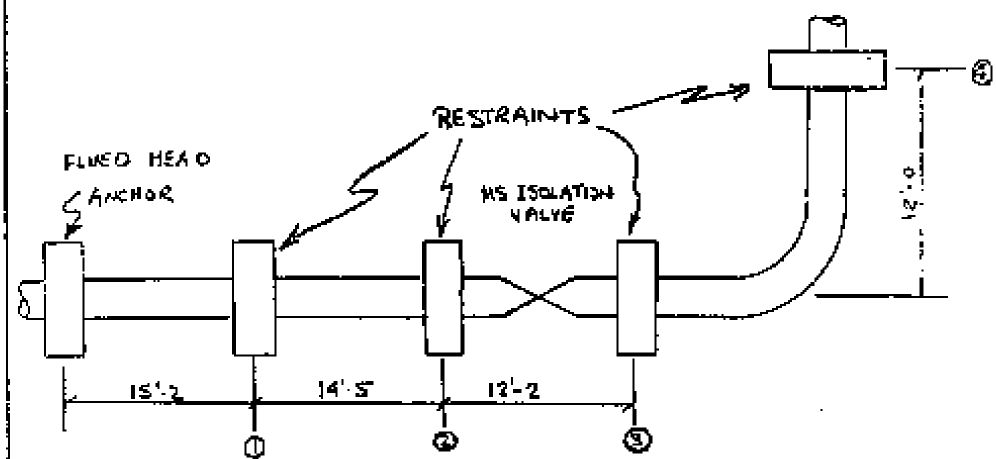
10. The consequences of a main steam or feedwater line rupture in the seismic Class I portion of the line is discussed and analyzed in Sections 5, 6 and 7 of this report. Failure of the non-seismic Class I portions of these lines, the major portions of which are run through the turbine building, can not adversely affect the mitigation of the consequences of the accidents and the capability to bring the unit to a cold shutdown condition since there is no safety related equipment located in the turbine building. A failure of any non-seismic Class I portion of the main steam or feedwater lines located outside the turbine building can only lead to a condition less intense than that described in Section 5.

11. A steam line or feedwater line break will not directly or indirectly result in loss of redundancy of any portion of the protection system (as defined in IEEE-279), Class 1E electric system (as defined in IEEE-308), engineered safety feature equipment, cable penetrations, or their interconnecting cables required to mitigate the consequences of the accident and place the reactor in a cold shutdown condition.

    Some safety related cables that will experience a change in pressure and temperature conditions are those associated with the auxiliary feedwater system. All cable in the trestle area is routed through underground or above ground conduit which will act as a shield from the effects of a pipe rupture accident. The cables are fully enclosed and are thermally rated for a temperature of 90C which is below the steam escape temperature of 320F. Note, however, that the 32OF temperature is predicated on an adiabatic expansion, the duration of which is between 60 and 95 seconds, and that the extreme temperature will only be experienced by the pump towards which the slot break (equivalent

TABLE 1

STEAM TRESTLE RESTRAINT LOADS – MAIN STEAM

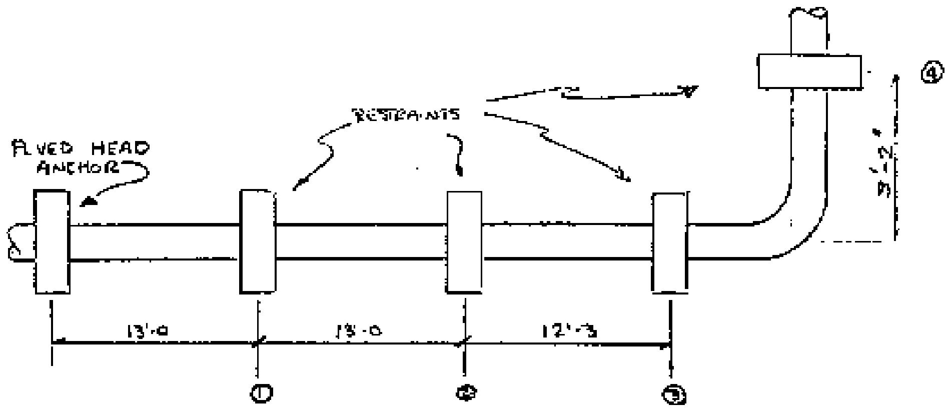


RESTR. NO.	PIPE RUPTURE AXIAL KIPS	PIPE RUPTURE TRANS KIPS	PIPE RUPTURE VERTICAL KIPS	THERMAL AXIAL KIPS	THERMAL TRANS KIPS	THERMAL VERTICAL KIPS	SEISMIC (DBE) AXIAL KIPS	SEISMIC (DBE) TRANS KIPS	SEISMIC (DBE) VERTICAL KIPS
F.H.	306	191	191	40	–	–	160	120	120
1	–	191	191	–	3	8	–	12	–
2	–	191	191	–	13	37	–	12	–
3	–	191	191	–	9	28	–	12	–
4	–	191	–	–	8	–	–	50	–

FORM ... REV ...

3C-6

STEAM TRESTLE RESTRAINT LOADS - FEEDWATER



RESTR. NO.	PIPE RUPTURE			THERMAL			SEISMIC (DBE)		
	AXIAL KIPS	TRANS KIPS	VERTICAL KIPS	AXIAL KIPS	TRANS KIPS	VERTICAL KIPS	AXIAL KIPS	TRANS KIPS	VERTICAL KIPS
FH	-	26	26	19	60	60	50	38	38
1	-	26	26	-	2	6	-	4	.
2	-	26	26	-	10	35	-	4	-
3	-	26	26	-	8	25	-	4	-
4	-	26		-	8	-	-	15	-

FORM 891 REV 7-71

3C-7

in area to cross-sectional area of pipe) jet is directed. It is expected that the cables associated with the impinged upon pump will suffer no adverse effects causing loss of function. Even if one pump is rendered inoperable, the two remaining auxiliary feedwater pumps have sufficient capacity to allow reactor cooldown to 300F.

Other electrical equipment expected to remain operable after the accident are the main steam isolation valves, the atmospheric dump valve and the steam line safety relief valves of the intact main steam line. The maximum expected impingement pressure on any conduit on the intact trestle is 15 psi.

12. The north wall of the control room is approximately 100 ft from the closest main steam line and approximately 85 ft from the closest feedwater line. Since both the MS and FW lines are located outside in the plant yard, no appreciable temperature buildup is anticipated. However, redundant control room air intakes are provided on the north and the south walls of the reactor auxiliary building. In the event that a temperature buildup occurs, air intake to the control room can be effected through the south wall intake. Another alternative would be to close both air intakes and run the ventilation system in the recirculation mode for as long as is necessary.

13. The safety related electrical equipment which could experience changes in temperature and pressure due to a pipe rupture accident are listed below:

    a. The motors of auxiliary feedwater pumps 1A and 1B

    b. The motor driven steam stop valve, I-MV-08-3, for the turbine driven auxiliary feedwater pump 1C

    c. The actuating circuitry of the main steam isolation valve (and seat bypass valve) of the intact system

    The manufacturers of the equipment listed above have stated that there should be no degradation of equipment due to the environment postulated to exist after a main steam or feedwater line rupture. All of the above listed equipment is designed for outdoor service and is expected to withstand the relatively short lived temperature transient resulting from a main steam or feedwater line rupture. All cables routed to this equipment are completely enclosed by conduit and are rated for 90C service. There will be no loss of system function if one auxiliary feedwater pump becomes inoperable; there will be no loss of system function if the motor driven steam stop valve (I-MV-08-3) becomes inoperable; and, there will be no loss of isolation function on the intact main steam line since the valve is designed to fail closed on loss of power.

Safety related features of the plant, other than those identified above, will not be affected by a main steam or feedwater line break since the propensity for damage is attenuated with distance and no additional safety related equipment is located in the area.

14. Drawing 8770-G-149, sheets 1 and 2, and drawing 8770-G-408, sheets 2A & 2B show the routing of the MS and FW lines from the containment to the turbine building, the locations of the auxiliary feedwater pumps and their respective piping, and the electrical cable routing in the trestle area.

    As stated in item 12 of this report, the control room wall and one of its 100 percent capacity ventilation intakes are approximately 85 ft south of the closest FW line and 100 ft south of the closest MS line. A second 100 percent capacity ventilation intake is located on the south side of the reactor auxiliary building. Both ventilation intakes are located at elevation +78' 9".

15. As stated in section 5 of this report, there is no potential for a flooding event caused by a main steam or feedwater line to prohibit the auxiliary feedwater system from performing its function. Each of the three pumps are surrounded by a flood wall to elevation +22 ft with an access opening which would preclude any water buildup.

16. The main steam and feedwater piping are designed as seismic Class I and nuclear safety class 2 (quality group B) from the steam generators up to and including the isolation valves outside the containment on the trestle. For fabrication and installation, these lines have complied with (fabrication) and will comply with (installation) the quality assurance requirements of class 2 piping as specified in ANSI B31.7. This entails 100 percent radiography of all welds as well as the non-destructive testing requirements stated in the material specifications. The St. Lucie Plant has an inservice inspection program for quality group B components.

17. Not applicable.

18. A full description of a postulated main steam line rupture and a Postulated feedwater line rupture is described in Sections 15.4.6 and 15.2.8 respectively.

19. A description of the seismic classification and quality group classification of the main steam and feedwater piping is presented in section 16 of this report. There are no other lines containing high energy fluid penetrating the containment structure during normal operation.

20. Since the main steam and feedwater system piping is not routed through any enclosed areas after emerging from the containment structure, buildup of pressure in these areas is not anticipated. In a similar manner, the only temperature buildup expected will be in the immediate area of the ruptured line and along the path of its resultant jet. Based on an adiabatic expansion of the steam line fluid, a very conservative jet temperature of 320F is assumed for the duration of the blowdown. The conservatism is manifest since no mixing or heat dissipation is considered in the assumption.

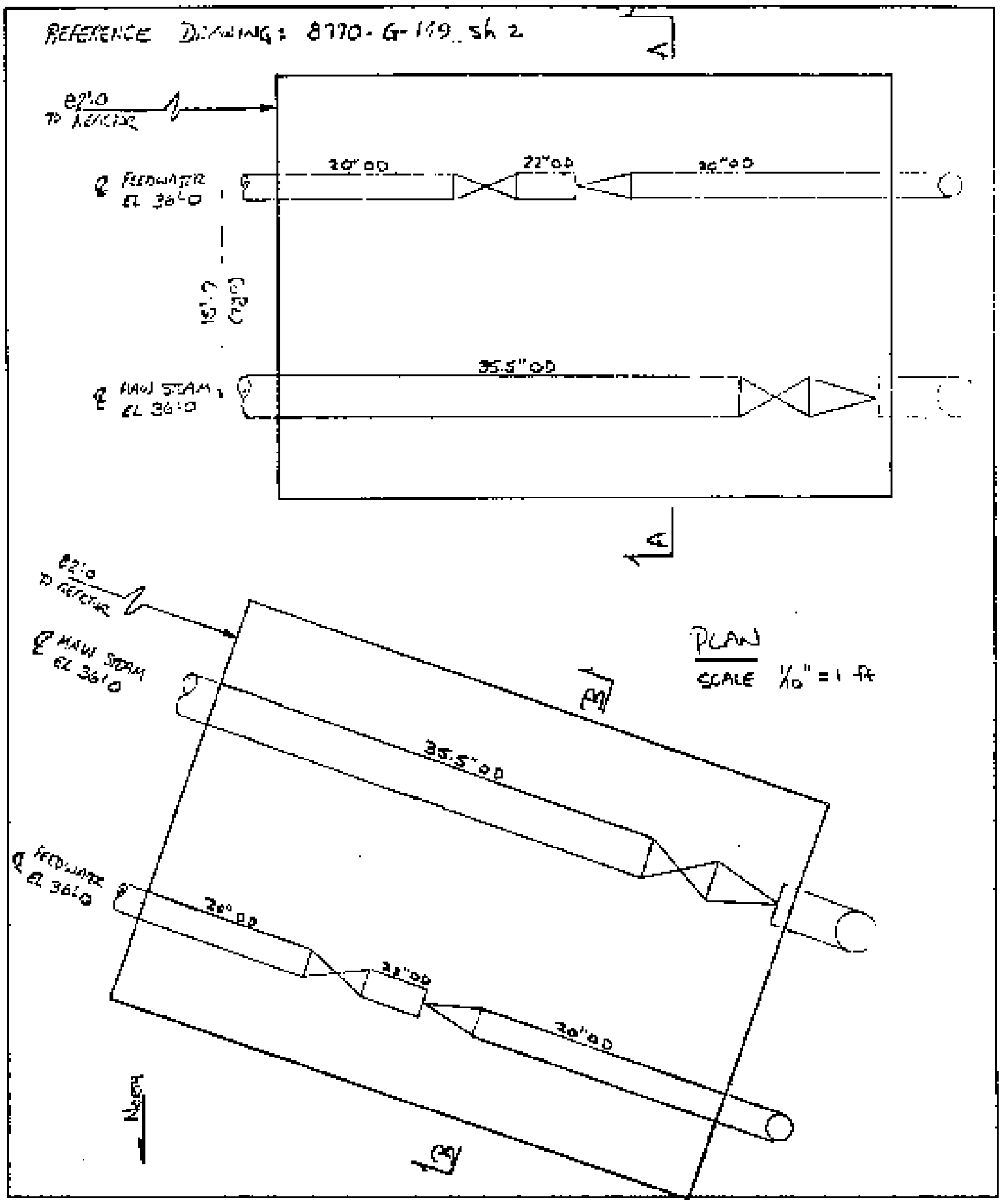
    Assumptions, methods, and results of analyses concerning the steam generator blowdown is presented in Section 15.4.6 of the St. Lucie Unit 1 FSAR.

21. Any pipe rupture in either the MS or FW system outside containment is not expected to damage the containment structure based on the following reasons:

    a. As described in section 7 of this report, the flued head anchor of the penetration is designed to accept the expected loads imparted from a complete pipe severance.

    b. The pipe whip restraints have been designed and spaced to prevent development of a plastic hinge.

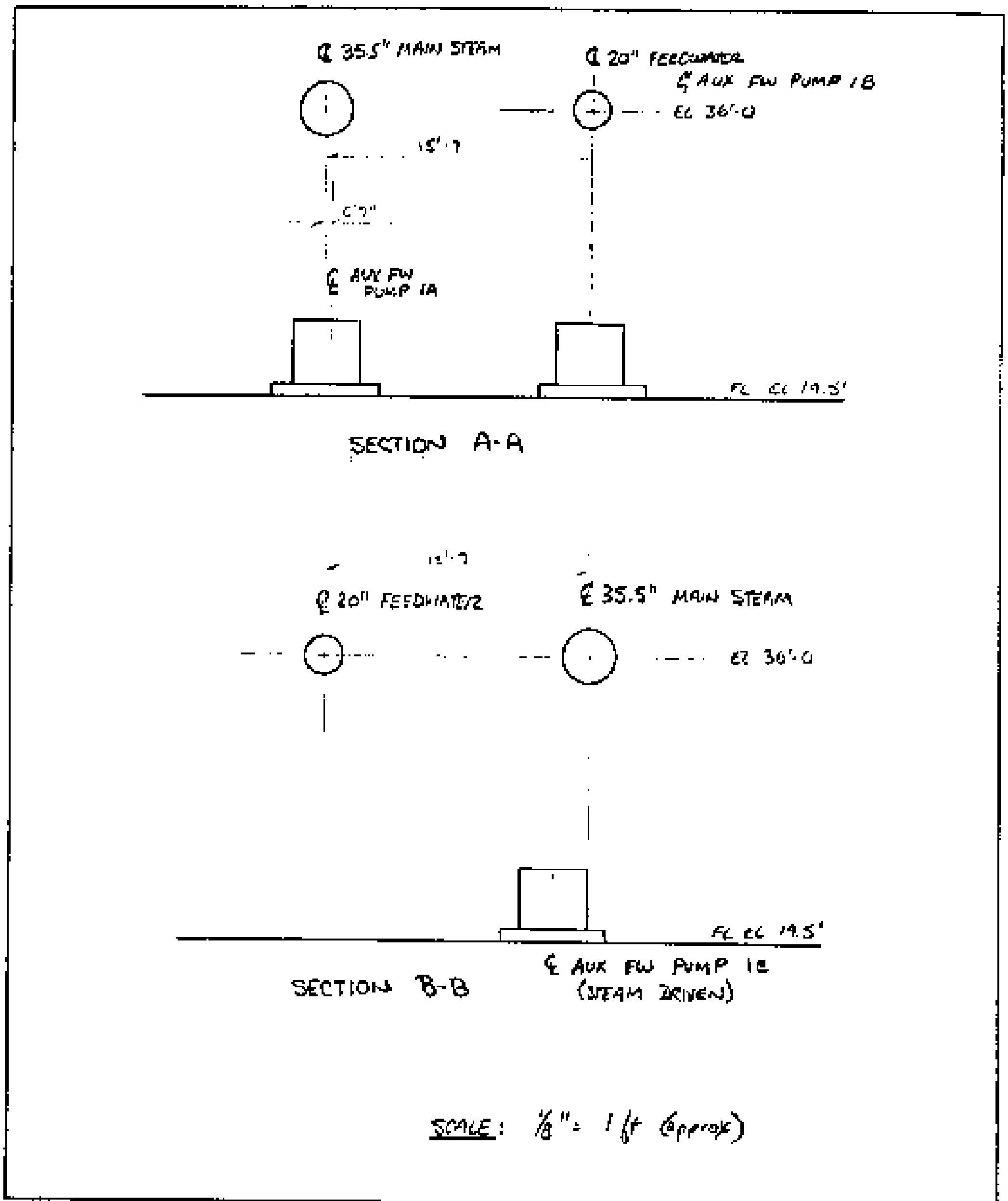
STEAM TRESTLE - STEAM & FEEDWATER PIPE ORIENTATION



FORM 891 REV 7-71

3C-11

STEAM TRESTLE - AUXILIARY FEED PUMP ORIENTATION



3C-12

APPENDIX 3D

ANALYSIS OF HIGH ENERGY LINE RUPTURE OUTSIDE CONTAINMENT*

INTRODUCTION

The systems analyzed herein are the lines used for shutdown cooling which includes portions of the low pressure safety injection system, chemical volume and control system (letdown and charging lines), steam generator blowdown system and auxiliary steam system.

In analyzing the effects of rupture in these high energy lines on systems or components required for safe shutdown, except as noted hereinafter, the following general criteria were considered:

a) For those concrete structures protecting systems and components essential to safe shutdown, the load combination of pipe rupture and design basis earthquake (DBE) is assumed

b) Single active failure in addition to the pipe rupture is assumed

c) Other than normal shutdown systems (e.g., ECCS) are considered acceptable to achieve safe shutdown

d) Piping which is pressurized only during testing is not considered

e) The criterion used to demonstrate structural adequacy is to show no loss of function.

* Refer to Appendix 3C for analysis of main steam and feedwater lines. Pursuant to the requirements of IE Bulletin 79-01B, a reevaluation of the environmental qualification of electrical equipment installed in the plant was performed. This updates a portion of the information provided in this appendix. See Section 3.11 for referencing to FPL responses to the bulletin.

Information provided in this appendix is historical and shall not be updated; however, it may still be similar to the re-evaluation documentation if no changes have taken place.

In addition to the MS and FW lines as presented in Appendix 3C, there are four pipe lines which have been identified as high energy pipelines outside the containment. These are the steam generator blowdown lines, auxiliary steam lines, letdown and charging lines, and the shutdown cooling system (including portions of the LPSI System). All these lines are located in or on the roof of the reactor auxiliary building which adjoins the containment structure.

These lines with the exception of the auxiliary steam lines are designed with steel pipe restraints which are supported from embedded plates in the concrete floors, walls or ceilings. The pipe restraints are only loaded in the event of a pipe break; separate pipe hangers or supports carry the normal pipe loads and separate seismic restraints provide seismic resistance.

Ultimately, all loads are resisted by the affected concrete walls, floors and ceilings. To analyze for the loads imposed on concrete elements, each affected pipe system has been traced so that all pipe loads imposed on a particular concrete element are identified and located. For pipe restraints, the pipe restraint which imposed the most critical load (in terms of magnitude and point of application) is applied to the concrete element. The concrete element is then analyzed in terms of all imposed loads in accordance with loading combinations indicated in Section 3.8.6.4.

It should be noted that the largest pipe break load for these lines imposed on a pipe restraint supported by concrete is in the order of 7,000 lbs. Normal pipe anchor loads are of similar magnitude. Most of the structural concrete walls and slabs affected by this investigation are 2 ft. thick with medium to heavy reinforcing. The loads imposed on the elements by piping, either normal or accident, are relatively small in comparison to the capacity of the element.

An analysis is presented herein of wall Number 5. This wall was identified as a critical element. As can be seen from the calculations, the flexural capacity of the wall is twice the imposed load. See Appendix 3E.

The effects of high energy pipe breaks have been reviewed for their effect on structural concrete elements in accordance with the criteria set in Section 3.8.6.4. No deficiencies in the structural strength of concrete elements have been determined; therefore, it is concluded that the effects of high energy pipe breaks outside the containment are within the capacity of the structural elements of the building.

ANALYSIS

The following analysis is based on the AEC issued "General Information Required for Consideration of the Effects of a Piping System Break Outside Containment."'

1) Protection against pipe whip is provided outside containment based on the following criteria:

   a) Shutdown Cooling System (including portions of the LPSI System)

      (1) Maximum operating pressure-and temperature during shutdown cooling is 450 psig and 300 F, respectively.

      (2) Although restrained, a ruptured shutdown cooling system line, were it not restrained, could impact a component cooling water line and a containment spray line. Of the two, the containment spray line is not necessary for safe shutdown.

   NOTE: The shutdown Cooling System entry temperature has been increased to 325F by revision to the plant Technical Specifications. This increase in temperature does not increase the potential for loss of structural function as detailed in this analysis.

   b) Chemical and Volume Control (CVCS) System - letdown and charging lines

      (1) Maximum operating pressure and temperature for the letdown line during normal operation is 2219 psia and 450F, respectively. Downstream of the letdown heat exchanger and after the pressure has been lowered by the letdown control valves, a pressure reducing valve lowers the pressure to 200 psig; temperature at this point is 140F.

      Maximum operating pressure and temperature for the charging line during normal operation is 2300 psig and 120F, respectively. However, any rupture in the line reduces the pressure to essentially zero since the charging pumps are of the reciprocating type.

      (2) Although restrained, a ruptured letdown line, were it not restrained upstream of the pressure reducing valve, could impact a component cooling water line which is necessary for safe shutdown.

   c) Steam Generator Blowdown System

      (1) Maximum operating pressure and temperature during normal operation is 900 psig and 532F, respectively.

      (2) Although restrained, a ruptured blowdown line were it not restrained could impact a component cooling water line which is necessary for safe shutdown.

d) Auxiliary Steam System

(1) Maximum operating pressure and temperature during normal operation is 40 psig and 350 F for one main section of the auxiliary steam supply system, and 75 psig and 350 F for the other. Various branch lines operate at lower pressures and temperatures and will be discussed separately.

(2) One branch of the auxiliary steam system supplies steam to the boric acid and waste concentrators and passes through portions of the RAB. Another branch supplies steam to the decontamination facility but does not enter the RAB except for the decon area itself.

2) The design criteria employed throughout the plant for the design of restraints and their spacing are predicated on restraining the individual pipe regardless of the location or orientation of all postulated ruptures. To achieve this, tables and charts were developed which indicate maximum spans allowable between restraints to prevent development of a plastic hinge and concomitant pipe whip. The tables and charts were prepared based on calculated load combinations which can be expected to result from either circumferential or longitudinal ruptures occurring along straight runs of piping, after elbows, etc. The tables and charts take into account variables such as size and configuration of the lines and the contained energy. Lines 1 inch in diameter or smaller are not considered.

3) Addressed in section 2 above.

4) A static analysis which combined the design loads of the piping at operating conditions, the loads calculated to exist after a pipe rupture, and DBE forces was utilized in the design of the pipe whip restraints and their spacing.

Both the circumferential and the longitudinal breaks considered are based on a flow area equivalent to the cross-sectional area of the pipe. Refer to Section 3.6 of the Safety Analysis Report for a complete description of pipe whip analysis method.

5) A description of the measures employed to protect against pipe whip, blowdown jet and reactive forces for the systems under consideration are presented in detail in Section 3.6 of the SAR. The criteria presented in Section 3.6 are extended to apply equally well to non-seismically designed portions of high energy piping. A summary of the criteria is presented below:

Pipe whip restraint locations for this plant are chosen based on maximum pipe spans required to develop ultimate moment and torque capabilities of the pipe cross section. Break locations in piping are assumed to occur at any location along the piping in all systems with normal operating pressures above 125 psig. The design loading combinations and stress criteria used for pipe restraint design are stated in SAR Section 3.6.3.

Actual pipe whip restraint location and spacing are shown on SAR Figures 3.6-10 and 3.6-19 through 3.6-32 for the shutdown cooling and low pressure safety injection system and on Figure 3D-1 for the letdown line. The auxiliary steam system is not restrained.

6) The evaluation of the structural adequacy of Category I structures as well as the design criteria used for these structures is discussed in Section 3.8.6.

7) The structural design loads, including the pressure and temperature transients, the dead, live and equipment loads, and the pipe and equipment static, thermal, and dynamic reactions are discussed in Section 3.8-6.

8) Seismic Category I structural elements such as floors, interior walls, exterior walls, building penetrations and the buildings as a whole have been analyzed for eventual reversal of loads due to the postulated accident. The accident forces are relatively low; design changes are not necessary, based on an analysis of all high energy lines.

   Ruptures of auxiliary steam lines will not adversely affect Category I structures,

9) Not applicable.

10) An evaluation of the consequences of a high energy line accident including failures caused by the accident is presented below:

    a) Shutdown Cooling System (including portions of LPSI system)

       The limiting or worst case shutdown cooling line rupture is the one which allows the greatest amount of reactor coolant to escape into the reactor auxiliary building, i.e., a rupture of a 12 inch line just at the onset of shutdown cooling. The temperature and pressure conditions at this time are 300 F and 450 psig, respectively. * The resultant temperature and pressure rise, 50 F and less than 1 psig respectively, have an insignificant effect on surrounding structures. Refer to Section 3.8.6.

See note in paragraph 1.a) for system max operating temperature revision.

b) Letdown Line

The limiting or worst case letdown line rupture occurs between the containment penetration and the pressure reducing valve where the internal energy is maximized. A guillotine rupture of the line is considered which results in a critical two phase blowdown into the reactor auxiliary building pipe tunnel area. The steam and water discharge at near sonic velocities through the 2 inch pipe.

A high temperature alarm downstream of the regenerative heat exchanger alerts the control room and also initiates closure of the upstream letdown isolation valves. Approximately 1000 lbs of letdown water will blowdown before valve closure. The water is assumed to be at 600 F since no cooling in the regenerative heat exchanger is expected at such a high flow rate. The resultant temperature and pressure rise, 50 F and less than 1 psig respectively, have an insignificant effect on surrounding structures. Refer to Section 3.8.6.

c) Steam Generator Blowdown Line

The letdown line rupture discussed in (b) above is limiting, i.e., the blowdown line rupture results in a less severe transient. The blowdown lines are isolated by one of two containment isolation valves (one inside and one outside of containment). Redundant primary devices detect a line rupture, alert the control room and initiate closure of the isolation valves.

d) Auxiliary Steam System

There will be no structural failures as a result of auxiliary steam line ruptures.

Auxiliary steam lines 3-AS-13, 14 and 16 (refer to Figure 3D-2) have been terminated (flanged off) prior to entrance into the RAB and as close to the main header as possible. Auxiliary steam lines 12-AS-1 and 3/4-AS-31 are the only auxiliary steam lines to enter the RAB, although only for a short distance and not near safety related equipment.
Nonetheless, redundant heat sensors have been added along the run to alert the control room of a rupture and to automatically terminate blowdown by closing valves PCV-16-1 and MV-08-12 for 12-AS-1, and valves PCV-16-6 and TCV-08-5 for 3/4-AS-31. (Refer to Figure 3D-2).

In addition, heat sensors have been added to the control room north outside air intake duct, which will automatically close the control room isolation valves in the event of a rupture of 12-AS-1 which is routed approximately 12 feet away. Since the line pressure at this point is only = 27 psig, impingement is not a concern. Calculations have shown that the maximum temperature rise in the control room is within 20 F of ambient assuming 750 cfm (intake fan capacity) of steam is drawn directly into the control room until closure of the isolation valves.

11) The analysis presented below is an evaluation of the effect of high energy line ruptures on safety-related systems necessary to mitigate the consequences of those ruptures and place the reactor in a cold shutdown condition.

Safety related systems needed to mitigate the effects of any particular high energy line break may vary depending on the particular circumstances and assumptions. These are discussed on a case by case basis below. Safety related systems needed to bring the reactor to a cold shutdown generally include the auxiliary feedwater system (assuming main feedwater is not available) and the shutdown cooling system (including portions of the LPSI system).

Further, the analysis investigates cases where high energy line ruptures may result in environmentally induced failures in other protection systems which are not required to function to mitigate the rupture or bring the plant to cold shutdown. For these cases loss of redundancy but no loss of function is permitted.

a) Shutdown Cooling System (including portions of the LPSI system)

A shutdown cooling line rupture does not initiate an automatic protective system function. Since the shutdown process is a carefully controlled administrative process, the operator is always aware of critical system parameters as pointed out in Section 20 of this appendix.

During the shutdown cooling mode of operation, the shutdown cooling system can accept a rupture of one shutdown cooling line and a single active failure. However, it is not a design basis of the system to accept any passive failure although selective passive failures could be tolerated.

Although not designed for passive failures, a single pipe rupture in the shutdown cooling system can not affect other portions of the shutdown cooling system which are redundant. This is accomplished by physical separation. However, certain portions of the shutdown cooling system which are not redundant could disrupt the shutdown cooling process were they to rupture.

The calculated temperature rise to 170F will not have any effect on cables in the area since these cables are qualified to 270F, 44 psig and 100 percent humidity for at least 15 minutes, for longer than the duration of the blowdown effects. All electric pump motors and motor operators for valves are designed for a 40C temperature rise.

b) Letdown Line

A letdown line rupture could initiate an ESFAS signal (SIAS) if reactor coolant pressure dropped below the SIAS actuation setpoint. If an SIAS were generated, it would automatically close the letdown loop isolation valves thereby terminating blowdown. Blowdown may be terminated before the SIAS by the temperature element downstream of the regenerative heat exchanger which also automatically closes the isolation valve (on high temperature). Since the isolation valve is located inside containment, any rupture outside containment, will not effect them.

A letdown line rupture can not affect the operation of the auxiliary feedwater system or the shutdown cooling system due to physical separation of the components.

The calculated temperature rise to 175 F will not have any detrimental effect on cables in the area since these cables are qualified to 270 F, 44 psig and 100 percent humidity for at least 15 minutes, far longer than the duration of the blowdown effects.

c) Steam Generator Blowdown Line

A blowdown line rupture would not initiate an automatic protective system function. Flow through a ruptured blowdown line will be automatically terminated by primary devices located downstream of the isolation valves. The steam air environment resulting from line rupture will not adversely affect the sensors or isolation valve located outside of containment.

The effect of blowdown line rupture and subsequent temperature rise in the pipe tunnel is less severe than that calculated for the letdown line, therefore, no detrimental effect on cables in the area is expected.

d) Auxiliary Steam System

The only auxiliary steam lines entering the RAB are 12-AS-1 and 3/4-AS-31. Neither of the lines are routed near safety related equipment and by virtue of the relatively low line pressures, = 27 psig and 75 psig respectively, ruptures will have no effect on safety related structures. In any event, ruptures will be detected by local redundant heat sensors and automatically terminated.

12) Control room habitability is not affected by any high energy line break since redundant heat sensors automatically isolate the control room if unusually high temperatures are sensed.

13) The necessity for electrical equipment to operate as a result of high energy line breaks is discussed below:

All safety related electric pump motors and motor operators for valves are designed for a 40 C temperature rise as stated in Section 11. All electrical cable to safety related equipment outside containment is qualified for 270 F, 44 psig and 100 percent humidity for 15 minutes as stated in Section 11.

Since no protection system functions are necessary to mitigate the effects of ruptures in the high energy lines considered, no equipment within the sphere of influence of the rupture need operate. Note, however, that all safety related equipment in proximity to the ruptures considered are designed for environmental conditions more severe than those calculated to exist because of the ruptures. This assures no loss of redundancy even in protection systems not required to function as a result of the rupture.

Sufficient physical separation and pipe whip restraints preclude damage to protection system equipment from either jet impingement or pipe whip. No barriers are thus required.

As stated in Section 12, the control room and its equipment will not be affected by any high energy line rupture.

14) Design diagrams for the shutdown cooling and LPSI systems are provided in the SAR as Figures 3.6-10 and 3.6-19 through 3.6-32. The letdown line is shown on Figure 3D-1. A drawing showing the routing of the auxiliary steam system is shown on Figure 3D-2.

15) The potential for flooding of safety relates equipment in tile event of a high energy line rupture is discussed below:

a) Shutdown Cooling System (including portions of the LPSI system)

A rupture in any portion of the shutdown cooling or LPSI piping could directly or eventually drain to the ECCS pump rooms which house safety related equipment. However, these rooms are equipped with sumps which alarm in the control room on high level which allows the operator to isolate all drain lines to that room or to shut off the source of the leak or both.

b) Letdown, Blowdown and Auxiliary Steam Systems

There exists no potential for flooding safety related equipment since the volume of water involved in any case is limited. In any event, all leakage to the ECCS pump rooms is alarmed and isolable.

16) The quality control and inspection programs followed for piping systems outside the containment is as described below:

a) Shutdown Cooling System (including portions of the LPSI system)

For St. Lucie Unit 1, the system is designed as Quality Group A and B outside containment (St. Lucie Unit 2 system are Quality Group B). For fabrication and installation on Unit 1, these lines will comply with the quality assurance requirements of code class 1 and 2 piping as specified by ANSI B31.7 (for Unit 2 the ASME Section III code class 2 is used). This entails 100 percent radiography of all welds as well as the non-destructive testing requirements stated in the material specifications. These components are covered by the plant inservice inspection program.

b) Letdown System

Outside of containment, this line is Quality Group B up to the letdown control valves designed to code class 2 of ANSI B31.7 for Unit 1, and designed to code class 2 of ASME Section III for Unit 2. The quality control and inspection specified for class 2 applies.

c) Blowdown Line

Outside of containment up to and including the isolation valve, the blowdown line is Quality Group B designed and inspected as in item (b) above. In the original plant design, beyond the isolation valves up to and including the primary (pressure) sensors, the blowdown line was Quality Group C and seismic Category I. Beyond the sensors the line is designed to ANSI B31.1 standards. The NDT requirements are consistent with that required by the code. Since the line is at high pressure, at least 10 percent of the welds in the line are radiographed.

d) Auxiliary Steam Lines

These lines are designed to ANSI B31.1 standards and they do not have any special NDT beyond that required by the code.

17) Not applicable.

18) Since ruptures in the letdown, blowdown and auxiliary steam systems have no effect on the plant's ability to shut down, the emergency procedures described in SAR Section 9.3.6 for the shutdown cooling system apply for all cases.

19) High energy lines that pass near structures, systems or components important to safety are designed as seismic Class I and either Quality Group A, B or C, or the analysis of ruptures provided herein has shown no degradation in the plant's ability to either shutdown or mitigate an accident

20) The assumptions and methods used in the analyses follow:

a) Shutdown Cooling System (including portions of the LPSI system)

The system operates only during shutdown and the operator monitors the cooldown rate to assure compliance with the Technical Specifications by observing flows, pressures and temperatures. In addition a number of flow, temperature and pressure instrumentation is provided to indicate and/or alarm abnormal conditions in the control room. Typically these include:

1) FIC 3306 will indicate loss of flow and open FCV-3306

2) PI 3303 - will indicate loss of pressure on inlet to SDHX

3) TR 3351 - will indicate low temperature

4) PI 3307 - will indicate loss of pressure

5) PI 1102 - will alarm low level in pressurizer

6) Changing pump flow rate will increase

The ECCS rooms and pipe tunnel communicate with other parts of the reactor auxiliary building, i.e., it is not isolated. Certain equipment and structures in the ECCS room are available as heat sinks.

With the above assumptions, the calculated pressure rise from a rupture of a line in the pipe tunnel or ECCS room is below 1 psig. The temperature rise is approximately 50 F.

In order to terminate blowdown following a rupture, the operator has to close the shutdown cooling line isolation valves located inside containment. With two valves in each line, a single active failure will not preclude isolation.

b) Letdown Line

The assumptions used in calculating a pressure and temperature rise in the pipe tunnel for a letdown line rupture are:

1) Guillotine break with 146.7 lbs/sec blowdown

2) Closure of letdown line isolation valves by temperature element downstream of regenerative heat exchanger

3) Subsequent to the rupture, no credit is taken for temperature reduction through the regenerative heat exchanger.

4) Blowdown is terminated in approximately 6 seconds.

5) No significant venting area from the pipe tunnel

With the above assumptions, the calculated pressure rise from a rupture of a line in the pipe tunnel is below 1 psig. The temperature rises to approximately 175 F.

c) Blowdown Line

The assumptions used in calculating a pressure and temperature rise in the pipe tunnel are similar to item (b) above with the following exceptions:

1) Blowdown is at 900 psig and 532 F

2) Blowdown is terminated in approximately 15 seconds

The results are less severe than those indicated in item (b) above.

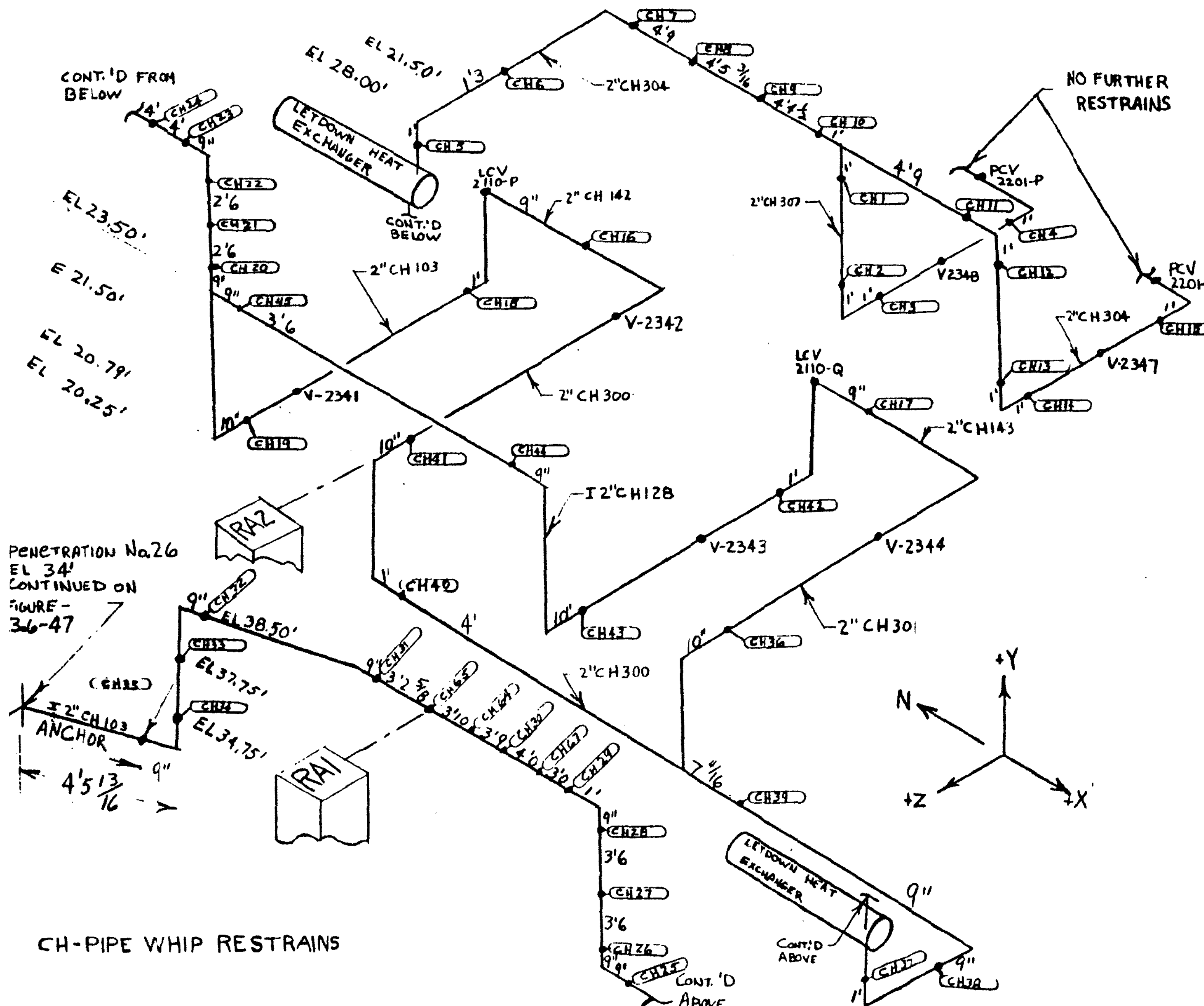
d) Auxiliary Steam System

Auxiliary steam lines in the RAB are not routed near safety related equipment. In addition 12-AS-1 does not pass through any enclosed areas so that structurally significant pressure buildup is not possible. In any event blowdown is terminated automatically if there is an abnormal increase in ambient temperature.

Outside of the RAB where 12-AS-1 passes approximately 12 feet from the control room north outside air intake, a conservative calculation assumed that 750 cfm of 500°F steam was drawn into the control room. In actuality the steam temperature is approximately 350°F in the line and the control room isolation valve would close well before 750 cfm could be drawn in. Even for this conservative set of assumptions, the control room temperature rose only 12°F above ambient.

Auxiliary steam line 3/4-AS-31 is not routed in any portion of the RAB except the decontamination area itself. The decontamination area by definition is a controlled atmosphere area exhausted through HEPA filters at a rate of approximately 1200 cfm, when decon is not in progress and 10,000 cfm when decon is in progress. A line rupture would trigger the heat sensors and initiate termination of steam blowdown in approximately 5 seconds. Assuming critical flow through the 3/4 inch line and a 1200 cfm atmosphere exhaust rate, calculations predict an insignificant pressure increase, i.e., less than 1 psig.

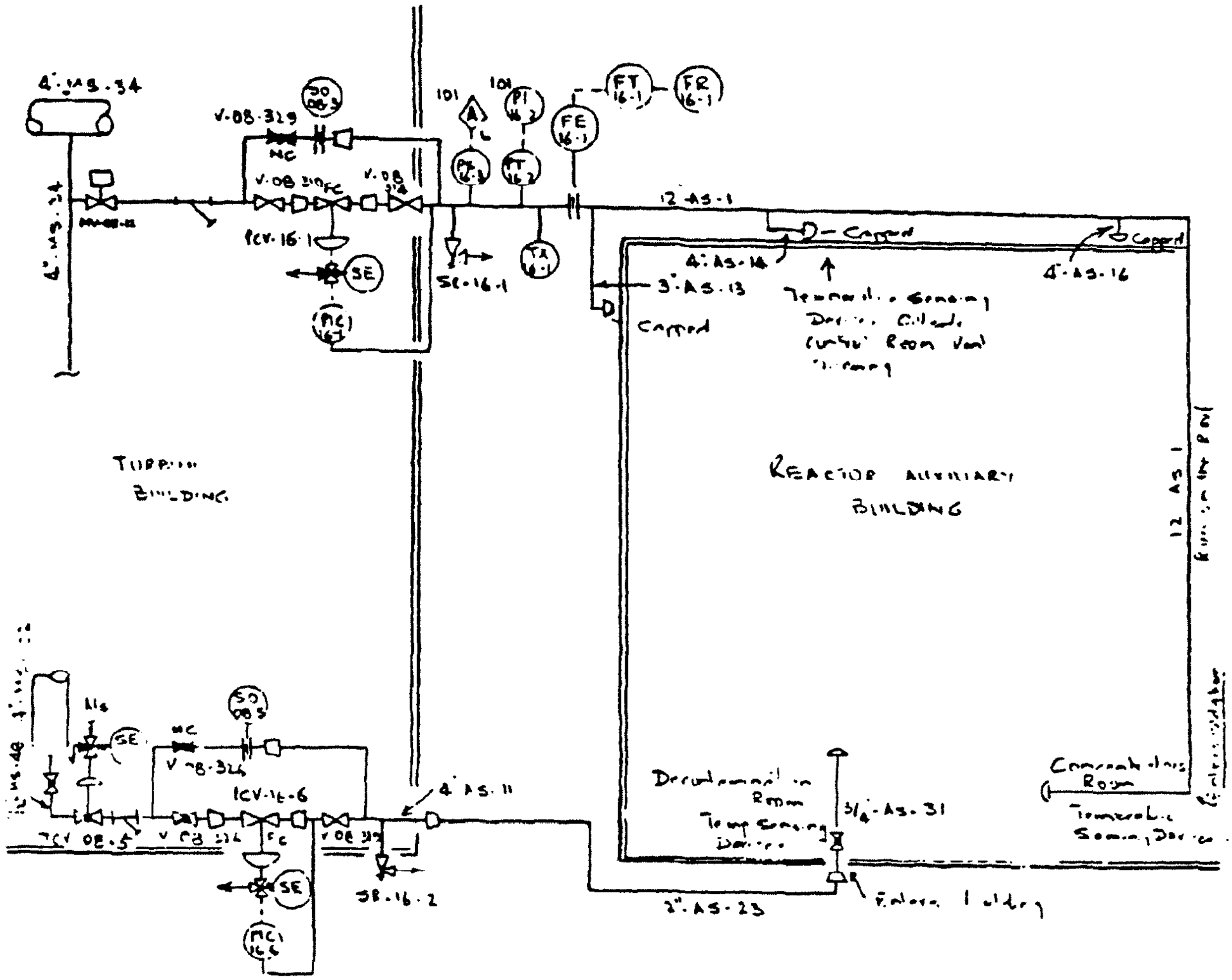
21) All penetrations through the primary and secondary containment are designed to accommodate pipe rupture loads. No significant loads from the rupture are transmitted to the secondary containment by virtue of the design of the penetration assemblies.



CH-PIPE WHIP RESTRAINS

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

CVCS PIPE WHIP
RESTRAINTS - RAB
FIGURE 3D-1



AMENDMENT NO. 12 (12/93)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT – UNIT 1

AUXILIARY STEAM LINE
RUPTURE INSIDE R, B

FIGURE 3D–2

APPENDIX 3E

R.A.B. INTERIOR WALL REVIEW

EBASCO SERVICES INCORPORATED

NEW YORK

BY AAB	DATE 5/29/74	SHEET 1 OF 3	
CHKD. BY A.G	DATE 5/30/74	OFS NO. 2998	DEPT. NO. 64

CLIENT F L O

PROJECT ST. LUCIE UNIT #1

SUBJECT R. A. B WALL AT EL 19.5' (WALL NO. 5)

FACTORED LOAD CONDITION

PIPE BREAK WITH DESIGN BASIS E.Q.

$$U = 1.0(L+D) + 1.0(R_a + T_a) + 1.0(Y_r + Y_j + Y_m) + 1.0(\underline{P}_a + F_{eqs})$$

WHERE:

$L = 0$

$D = 0$ (DEAD LOAD IS A VERTICAL [illegible])

$R_a =$ 50 PSF - VERTICAL REACTION ON WALL

$T_a = 0$

$Y_r =$ HIGH ENERGY PIPE BREAK REACTION *

$Y_j = 0$

$Y_m = 0$

$\underline{P}_a = 0$

$F_{eqs} =$ HORZ. D. B. E. LOAD (SSE)

$$\therefore U = 1.0(R_a + F_{eqs} + Y_r)$$

* 2 LOADS EACH 5.6^k MULTIPLIED BY DYNAMIC LOAD FACTOR OF 3 = 16.8^k

FORM 561 REV 7-71

BY AAB DATE 5/21/79

CHKD. BY AG DATE [illegible]

CLIENT FLO

PROJECT ST. LUCIE UNIT #1

SUBJECT R.A.B. WALL AT EL. 19.5'

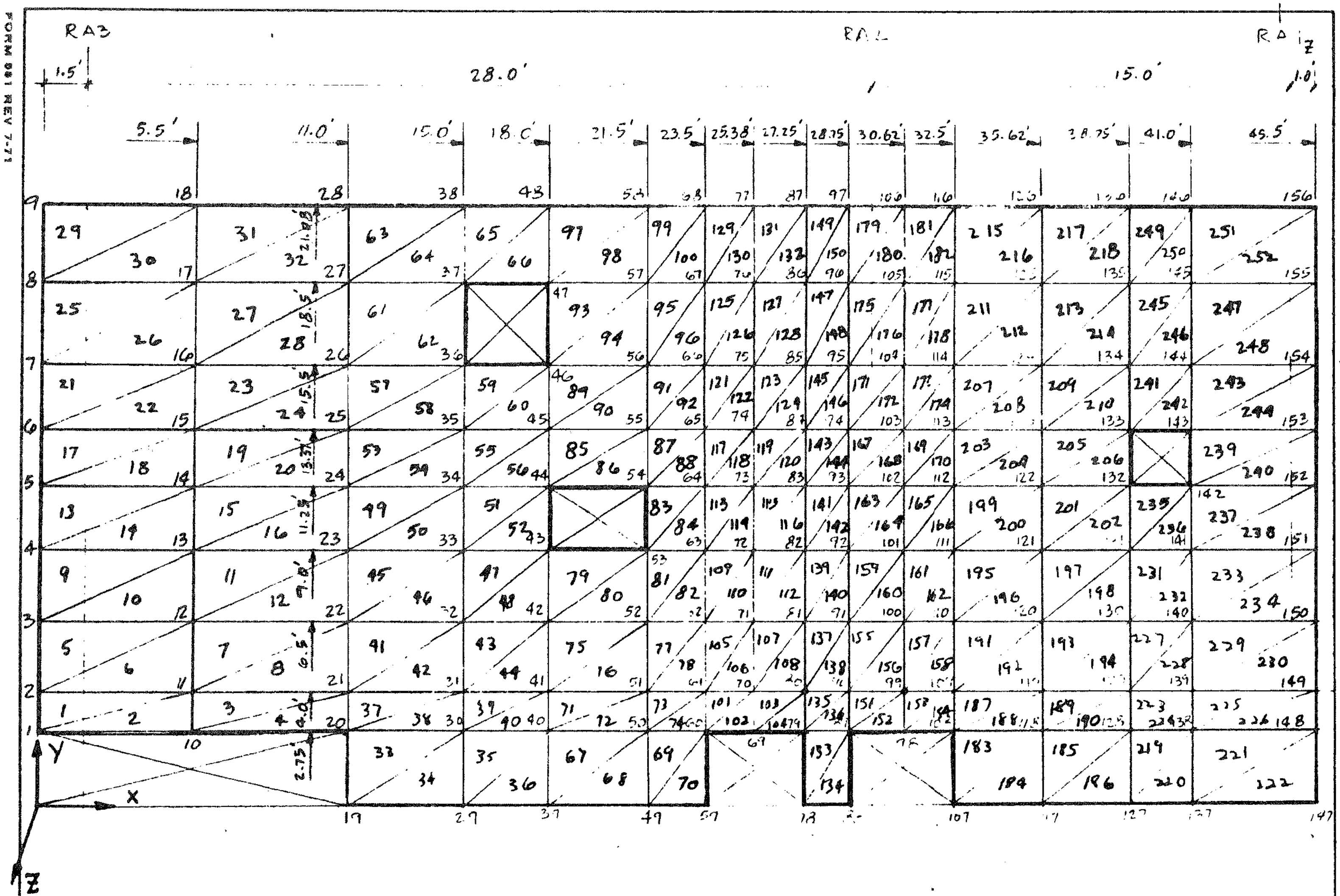
OFS NO. 2998

SHEET 2 OF 3

DEPT. NO. [illegible]

FORM 881 REV 7-71

3E-2



FINITE ELEMENT (PLATE) MODEL FOR WALL NO. 5

(SEE P. 1)

EBASCO SERVICES INCORPORATED

NEW YORK

BY AAB DATE 5/29/74 SHEET 3 OF 3

CHKD. BY A.G DATE 5/7/76 OFS NO. 2598 DEPT. NO. 64

CLIENT FLO

PROJECT ST. LUCIE UNIT #1

SUBJECT R.A.B. WALL AT EL. 19.5' - (WALL NO. 5)

CAPACITY OF WALL ELEMENTS

1. ELEMENT AT RIGHT OF COL LINE RA2 (AT OPENING) - VERTICAL

$t = 18''$, $d = 15''$, $b = 56''$

$\phi = 0.85$, $f'_c = 3$ ksi, $f_y = 40$ ksi

Reinf. #6@12" = 0.44 in²/' + 2-#6

$A_s = 0.44 \times 4.67 + 2 \times 0.44 = 2.93 \text{ in}^2$

$$M_u = \phi\left[A_s f_y \left(d - \frac{a}{2}\right)\right]$$

$$a = \frac{A_s f_y}{0.85 f'_c b} = \frac{2.93 \times 40}{0.85 \times 3 \times 56} = 0.82''$$

$$M_u = 0.9\left[2.93 \times 40\left(15.0 - \frac{0.82}{2}\right)\right] = 1540^{k\text{-}in}$$

or $M_u = 128^{k'}$

MOMENT DUE TO PIPE BREAK (NODES 107 & 117) AND SSE LOAD = $(25.0 + 9.4) + (9.1 + 9.1) = 52.6^{k'} < 128^{k'}$

~~8K IS MAX. WALL CAPACITY FOR U.S.D.~~ O.K

2. ELEMENT AT LEFT OF COL LINE RA2 (AT OPENING) - VERTICAL

$d = 15''$ $b = 45''$, #6@12"c + 2-#6

$A_s = 3.75 \times 0.44 + 2 \times 0.44 = 2.53 \text{ in}^2$

$$a = \frac{2.53 \times 40}{0.85 \times 3 \times 45} = 0.88''$$

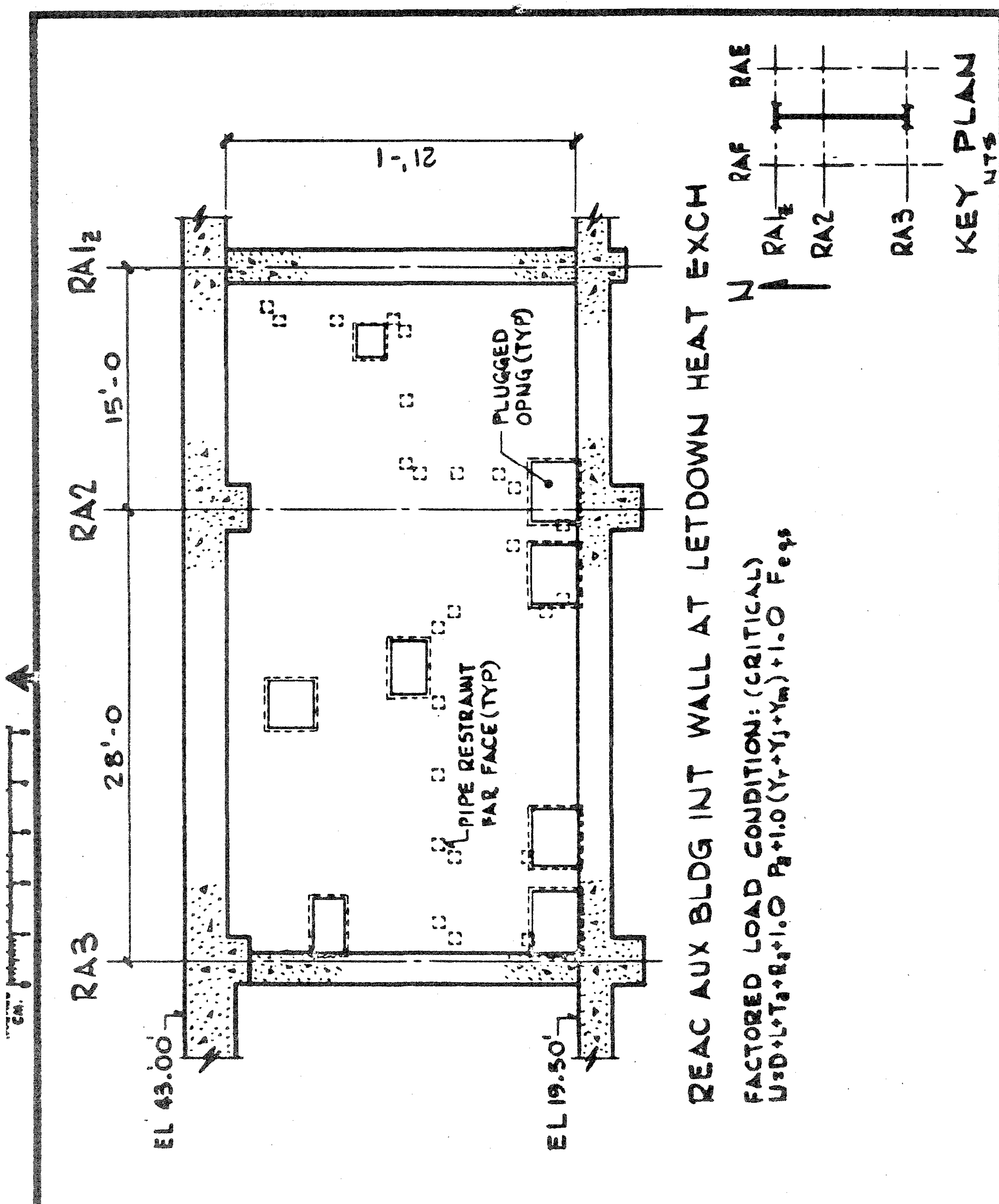
$$M_u = 0.9\left[2.53 \times 40\left(15.0 - \frac{0.88}{2}\right)\right] = 1326^{k\text{-}in}$$

$M_u = 110^{k'}$

MOMENT DUE TO PIPE BREAK (NODES 59 & 49) AND SSE LOAD $= (20.3 + 8.5) + (3.9 + 9.7) = 42.4^{k'} < 110^{k'}$

O.K

FORM 581 REV 7-71



EBASCO SERVICES INCORPORATED	FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT 1973-890,000 KW INSTALLATION-UNIT No. 1 HIGH ENERGY PIPE BREAK STUDY	A-
DIV. CH BY VJP SCALE NTS CH H.S. DATE APPROVED		

APPENDIX 3F

CAPABILITY OF FACILITY TO ACCOMMODATE TORNADO MISSILES

1.0 INTRODUCTION

St. Lucie Unit 1 facility has been designed to withstand the wind loading associated with tornadic winds of 360 mph. In addition, impact of either a 10' x 2" x 4" plank traveling at 360 mph (528 fps) or a 4000 pound automobile traveling at 50 mph (73 fps) has been accommodated by design. Implementation of these tornado design criteria is discussed in Sections 3.3.2 and 3.5.

Tornado protection of equipment required to achieve and maintain safe shutdown is provided by enclosing the equipment in protective structures, or by providing redundant components sufficiently separated to preclude interaction of both by a single design basis missile, i.e., 2" x 4" x 10' plank or automobile. The capability of the unit to accommodate the design basis tornado is further enhanced by the protection afforded by plant structures, and the inherent capability of components to accommodate a missile impact spectrum without loss of function. These inherent capabilities were not taken credit for in the implementation of the tornado criteria discussed in Sections 3.3.2 and 3.5.

The Regulatory Staff in its letter of August 26, 1974 requested that the capability of the facility be reviewed with regard to the tornado missiles specified by the Staff during its review. This review consisted of three fundamental phases, namely; a review of the characteristics of tornadoes indigenous to peninsular Florida; the development of a model to predict missile characteristics in tornadoes and the application of this model to the missile spectrum identified by the Staff; and a study of the capability of equipment not enclosed in protective structures to accommodate tornado generated missiles.

2.0 THE FLORIDA TORNADO

Section 2.3.1.3 (d) and Appendices 2C and 2F discuss tornado characteristics indigenous to peninsular Florida. Two independent studies, covering the 85 year period from 1887 to 1972, both conclude that tornadoes of historic record have had wind speeds of less than 200 mph and that the occurrence of a severe, characteristic of the midwest, tornado in Florida is unlikely. The latter conclusion is supported by meteorological considerations and the historic record.

Path length and width data show that the expected tornado contact area in peninsular Florida is more than a factor of 10 less than the 2.82 mi^2 determined by Thom (1963)[1] for Iowa. Howe (1974)[2] indicates a contact area of 0.17 mi^2 for Florida, which is consistent with the Dames and Moore value of 0.26 mi^2 (Appendix 2F).

Path size and intensity have been correlated by several investigators. Most recently, Wilson and Morgan (1971)[3] pointed out that (i) long path tornadoes cause over three times as many fatalities per mile as all other tornadoes, and (ii) they occur most frequently in the Mississippi Valley region during the spring months. Tornadoes are most frequent in Florida in the summer months of June, July, and August. Typically, the intense and long track tornadoes occur in the same geographic areas of the U.S.

By summer mid-level jet stream circulation (favorable to the formation of long-track, multiple outbreak, and other severe tornado events normally occurring in the spring) has weakened and is well to the north of Florida. Without the penetration of the jetstream, strong conditional instability and its subsequent trigger mechanism (such as a cold front) are absent over peninsular Florida in the summer. Thus, organized convection and probably the rotating cyclones which spawn severe tornadoes do not form during the period of maximum tornado activity.

Florida tornadoes show a marked peak of occurrence frequency in the mid afternoon, which is much sharper than in the central U.S. This characteristic is actually associated with the predominance of summertime tornadoes. It is a further suggestion that many of the Florida tornadoes occur with afternoon heating and sea breeze thundershowers which lack the upper level dynamic support for organized tornado formation. Some tornadoes are probably of waterspout origin as well as hurricane-spawned. The latter are also less intense than those originating from extratropical cyclonic storms.

The results of the statistical and engineering studies of Florida tornadoes (Appendices 2C and 2F) are of significance meteorologically. In the past two decades, great progress has been made both in the development of a body of tornado statistics as well as understanding the primary atmospheric conditions which spawn them. In the early to mid 1950's, various atmospheric conditions, identified in proximity radiosonde soundings and synoptic types, indicated that tornadoes originated under different conditions in different regions of the country (Fawbush and Miller[4], 1954; and Beebe[5], 1958). The general notion that tornadic intensity and frequency are a function of atomspheric conditions became evident, particularly to those involved in severe weather forecasting.

The midwest tornado frequency and intensity reflects the relative location of the Gulf of Mexico and the Rocky Mountains. The Gulf is the source of the moist air stream at low altitudes, whose large energy content (latent heat) can provide rapid ascent once a triggering mechanism provides a sufficient vertical displacement. The mountains impede low level westerly flow, but allow the westerly passage of dry (potentially cool) air at higher altitudes.

Cyclonic development frequently occurs in the lee of the Rocky Mountains. Interaction with the Gulf air can result in intensification of the westerly upper level flow into a principal jet stream. This is predominantly a mid latitude effect that occurs north of peninsular Florida. The occurrence of this type of synoptic condition over peninsular Florida is infrequent due to its distance from the Rocky Mountains. Thus, based on meteorological considerations the risk associated with severe tornado activity is much less in Florida than in regions to the north and west where atmospheric conditions favor severe tornado development. The historical record supports this phenomenological conclusion.

3.0 TORNADO MISSILES

Appendix 3E to the St. Lucie Unit 2 PSAR (docket 50-389, Amendment 21 dated October 11, 1974) provides a description of the model used to evaluate tornadic missile behavior as well as the results of studies of the missiles identified by the Staff. This study utilized the probabilities for geometric strike and wind speed developed in WASH-1300[6]. The WASH-1300 study analyzed 1612 tornadoes occurring in the continental U. S. during 1971 and 1972; developed a geometric strike probability based on path length and width characteristic of Iowa; and developed a windspeed probability technique based on Fujita's intensity scale. The Fujita scale results from a subjective evaluation of observed damage, and an intensity distribution derived for the entire U.S. (581 tornadoes of F2 intensity or higher). Although not considered germane to a peninsular Florida tornado, the WASH-1300 methodology was used (in compliance with Regulatory requirements) to arrive at the probability of a missile attaining specific velocities in a tornado. For the upper limit, bounding tornado of historic record (200 mph), i.e., a tornado more intense than any of historic record:

a) Probability of a missile achieving its mean velocity is 1.5×10^{-5}

b) Probability of a missile achieving its maximum velocity is 7.5×10^{-9}

These probabilities assume a geometric strike probability for Iowa of 1.5×10^{-3}. Appendix 2F evaluates the geometric strike based on the Florida contact area, which is about an order of magnitude less than the Iowa contact area. The geometric strike probability for Florida's east coast is 5.17×10^{-4}. From Figure 2 of Appendix 2F the probability of achieving a 200 mph windspeed is about 2×10^{-3}. Thus, based on the most recent peninsular Florida tornado evaluation, the 200 mph tornado probabilities are more appropriately represented by:

a) Probability of a missile achieving its mean velocity is 5.2×10^{-7}

b) Probability of a missile achieving its maximum velocity is 2.6×10^{-10}

These latter probabilities, which are based on the Dames and Moore D&M study (Appendix 2F) are considered representative of Florida tornadoes at the St. Lucie site because:

a) The D&M study of 116 Florida coastal tornadoes from 1950 to 1972 reaches the same conclusion with regard to the intensity of Florida tornadoes as the earlier study of Professor Brooks, et. al., (Appendix 2C). The earlier study covered 429 Florida tornadoes from 1887 to 1968.

b) The contact area used for Florida is in agreement with that determined by Howe[2] in 1974. Thus, the smaller contact area for Florida tornadoes provides a more appropriate geometric strike probability.

c) The D&M intensity classification is based on an engineering evaluation of the effects of winds on structures, whereas, the Fujita scale is based on a subjective evaluation of observed damage.

d) The probability of achieving a given windspeed is based on 22 years of Florida coastal tornado data, whereas, the WASH-1300 value is based on 2 years of continental U. S. data.

It is recognized that at the time of this writing that the D&M intensity classification has not been published. However, it must be noted that this intensity classification has been developed under contract for the U.S.A.E.C's Division of Reactor Research and Development and is under review therein. It is anticipated that the D&M meteorological and engineering approach will become available in the literature subsequent to completion of RR&D's review.

Regardless of the technique used at arriving at the probability of achieving mean or maximum missile velocities, the occurrence of such missiles is extremely remote. It must also be noted that these probabilities are the product of three individual probabilities, namely,

P_1 - probability that a tornado will strike the site.
P_2 - probability that a tornado will attain winds of 200 mph.
P_3 - probability that a missile will be generated.

Calculation of P_3 is somewhat complex and is itself dependent on a number of factors which include the probability that an object which could become a potential missile will exist within the tornado path and, if so, the orientation of the object with respect to wind direction is amenable to missile development. Since no satisfactory method exists at present to calculate P_3, it has been conservatively assumed in the discussion supra that $P^3 = 1$, although for certain missiles, it is expected to be considerably less.

In addition to the above probabilities the following must also be considered in the evaluation of tornado missiles:

P_4 - probability that a missile, if generated, will impact the component.
P_5 - probability that, if struck, loss of system function occurs.
P_6 - probability that an independent single failure occurs in the struck component's redundant counterpart.

Once generated, there is some probability (P_4) that the missile will impact (interact) with the component (target) in question. This interaction may occur in two ways, namely; (a) a missile may impact with a given target as a result of its ejection from the tornado windfield at some distance from the target, in which case the trajectory of the missile upon leaving the windfield must be such that it will interact with the target, or (b) the missile can interact with the target within the windfield as both target and missile pass through the windfield.

Both types of interaction have been investigated heretofore to determine the maximum value of P_4, e.g., see Amendment 29 to Shearon Harris Units 1, 2, 3, 4, Dockets 50-400, 401, 402 and 403, at Section 9.8.6. Values of P_4 calculated for Shearon Harris ranged from 9.4×10^{-5} to 5.3×10^{-3} for missile impact on a service water pump. For purposes of this discussion, a value of 10^{-2} would provide a reasonable estimate of P_4 for St. Lucie Unit 1.

The capability of components to accommodate a strike without loss of the system's function (P_5) is difficult to specify quantitatively. A discussion of the inherent capability of vital components is discussed infra. For purposes of this discussion, $P_5 = 1$ is assumed. For P_6 a reliability level of 10^{-2} is appropriate.

The foregoing probabilities can be combined to assess the probability associated with design basis type missiles generated by the bounding historic tornado to cause concurrent loss of a component function and its redundant counterpart. These values are less than the following:

	WASH-1300 <u>Methodology</u>	Methodology appropriate <u>to Florida (D&M)</u>
Missile at mean velocity	1.5×10^{-9}	5.2×10^{-11}
Missile at maximum velocity	7.5×10^{-13}	2.6×10^{-14}

The probability of missile strike resulting in loss of single component function without loss of its redundant components function is 10^2 greater than these values, i.e., less than:

	WASH-1300 <u>Methodology</u>	Methodology appropriate <u>to Florida (D&M)</u>
Missile at mean velocity	1.5×10^{-7}	5.2×10^{-9}
Missile at maximum velocity	7.5×10^{-11}	2.6×10^{-12}

Regardless of the methodology, the probabilities associated with an unacceptable strike of a design basis type missile in a 200 mph tornado, assuming it can generate the missile, are acceptably low and still lower for the Staff's 360 mph tornado. The design of Unit 1 with regard to tornado strike is considered adequate for the large missiles assumed in the design or those identified by the Staff.

4.0 TORNADO GENERATED DEBRIS

Reassessment of the tornado design concept requested by the Staff reaffirmed the propriety of the design basis missiles and those identified by the Staff during its review. This notwithstanding, the evaluation of tornadic capability was extended to consider a spectrum of small objects (debris) that could be generated as a result of tornadic activity at the site and would undoubtedly be present in the tornado windfield. Although not very penetrating, i.e., their damage potential is minimal, their multiplicity potential need be addressed. Thus, the intent of the tornadic debris review was to assess the vulnerability of outdoor components to damage from these small miscellaneous objects that might be prevalent in a tornado windfield.

The likelihood of component damage from debris is also very small, i.e., the probabilities are acceptably low. However, the possibility for enhanced tornado resistance capability resulting from modest facility modifications was considered sufficient to merit consideration. This action should not be interpreted to mean that the capability of the facility to accommodate tornadic events is unacceptable. The design In this regard is adequate. Rather, any modifications defined infra are in accordance with the guidance provided by the Commission at 10 CFR 50.109, namely, backfitting is appropriate if "such action will provide substantial, additional protection." This basic criterion governs the considerations that follow.

4.1 EXPOSED SAFETY RELATED COMPONENTS

Installation of tornado missile protection for the: a) intake cooling water pumps, b) component cooling water pumps, c) diesel generator air intakes and exhausts, d) diesel generator access doors, e) diesel generator fuel oil pumps, and f) auxiliary feedwater pumps has been completed and condition of license item A deleted by Amendment No. 4 to license issued April 16, 1976. Therefore, commitments for additional missile shielding made in this Appendix are satisfied.

The following safety related components were not originally housed in Category I structures or otherwise specifically protected from direct tornado missile impact:

a) Intake Cooling Water System

1) Pumps
2) Piping and Valves (at intake structure and component cooling area)

b) Component Cooling System

1) Pumps
2) Heat Exchangers
3) Piping and Valves (in the heat exchanger area)

c) Diesel Generator Building Openings

1) D G Radiator Louvers
2) D G Ventilation Intake Louvers

3) D-G Ventilation Exhaust Louvers
4) Access Openings

d) Diesel Oil Fuel Storage and Transfer System

1) Storage Tanks
2) D.O. Transfer Pumps
3) Piping and Valves (between storage tanks and pumps).

e) Auxiliary Feedwater System

1) Condensate Storage Tank (top only)
2) Pumps
3) Piping and Valves (between pumps and main feedwater lines and main steam line)

f) Main Steam and Feedwater System

1) Piping and Valves (between containment and isolation valves)

g) Control Room Air Conditioning System

1) Chiller Units
2) Piping and Valves (to Chiller Units)

4.2 TORNADIC DEBRIS EVALUATION CRITERIA

In evaluating the effects of tornado missile impact on the exposed components listed in Section 4.1 the following criteria were applied to determine whether enhancement of tornado protection should be considered:

a) If it could be demonstrated that loss of function of the exposed component(s) would not prevent safe shutdown, no further tornado missile protection was considered necessary.

b) If it could be shown that the exposed component(s) has considerable inherent capability to resist missile penetration without loss of function, no further missile protection was considered necessary.

c) If it could be shown that a substantial level of protection is afforded the exposed component(s) by existing structures and components, no further missile protection was considered necessary.

d) If it could be shown that, in the event of damage to the exposed component(s) alternate protected systems or components would be capable of performing the function of damaged component(s) in achieving safe shutdown, no further missile protection was considered necessary for the exposed component(s).

If application of criteria (a) thru (d) or some combination thereof did not result in an acceptable conclusion, evaluation of an enhanced protection level was initiated. Commensurate with the guidance provided by 10 CFR 50.109 this evaluation struck an appropriate balance between the benefit afforded, i.e., the appropriate degree of enhanced protection; the feasibility of modifying existing structures; and the schedule for operability of the unit.

Enhanced protective structures, where provided, must be able to accommodate the wind loadings associated with current midwest type design basis tornado, i.e., 360 mph, and must accommodate the DBE without adversely affecting safety related components and structures. In addition, these structures must resist penetration of tornadic debris. A more than sufficient level of such protection is provided by 1.0 feet of concrete, or 1.0 inch of steel, or an equivalent combination thereof. The capability of this level of protection is readily demonstrated by considering the ability to accommodate penetration of the large design basis type missiles. Table 1 provides this data along with the maximum and mean velocities these missiles could achieve in a 200 mph tornado. It shows that 1.0 feet of reinforced concrete or its equivalent provides penetration resistance sufficient to accommodate the maximum velocities attained by the large missiles in a 200 mph tornado.

4.3 EVALUATION OF EXPOSED SAFETY RELATED COMPONENTS TO ACCOMMODATE TORNADIC GENERATED DEBRIS

The capability to accommodate tornadic debris has been evaluated for the safety related components identified in Section 4.1. The following paragraphs discuss this evaluation and with additional justification in Section 6.0. Figure 1.2-2 provides a site plot plan illustrating the relationship of the components and structures discussed below, and Figures 3F-1 to 3F-5 provide recent aerial views of the facility.

Components potentially vulnerable to damage from tornadic debris are the active ones, i.e., valve operators and pump motors. On the other hand, piping, pump motor casings and valve bodies, i.e., components comprising the pressure boundary of fluid systems, are essentially invulnerable to the spectrum of small light objects that are appropriately characterized as debris. It is for this reason that the discussion infra focuses on active components.

4.3.1 Intake Cooling Water (ICW) System

The intake structure is situated due west of the turbine building with little protection afforded by adjacent structures. (Figure 9.2-1a provides the ICW system P&ID.) ICW components located on this structure relevant to this discussion of tornado debris are the three ICW pump motors. Piping and valves associated with this system are located below grade and in the valve pit area immediately north of and adjacent to the pump area. Active components in the valve pit are the motor operators associated with valves MV-21-2 and MV-21-3.

Two of the valve operators in the valve pit are potentially vulnerable to vertical missiles, i.e., debris ejected from the tornado following a trajectory with a vertical component. (Figures 10a to 10c of the reference 7 illustrate typical trajectories). These are the operators associated with valves I-MV-21-2 and -3. The normal position of these valves is open and should isolation from the turbine cooling water system be necessary, the valves are closed from the control room. Position indication (red and green lights) is provided in the control room. Only one valve need be closed to insure adequate ICW flow. Should one or both valves become inoperable, manual valves SB-21213 and SB-21167 can be closed. Since the valve operators are located below grade, are physically well separated, have an alternate means available to isolate the tie lines to the turbine water cooling system and have redundant systems available, it is concluded that adequate tornado protection has been provided.

The motors of the ICW system utilize the concept of separation to achieve an acceptable level of tornado protection. The inherent capability of the motors to accommodate a spectrum of debris has not been addressed specifically. However, an appreciable capability over a broad range of the potential missile spectrum is not anticipated. Since the addition of a modest amount of missile protection for these motors was found to enhance the tornado debris capability of the ICW system, physical motor protection is provided. The motor protection allows for adequate natural cooling of the motors, accommodates maintenance requirements, and is compatible with the intake crane lift. The design provisions to accommodate maintenance include a removable structure for each ICW pump motor. Removal of the protective structure associated with a single pump allows access to the motor and pump for maintenance. Removal of the shield for maintenance does not affect the operability of the adjacent pump(s) provided that the missile shield for each in service pump is in place.

Intake cooling water system piping and valves are also located in the component cooling water area. Except for the connection to the CCW heat exchangers, the piping is not subject to impingement by horizontal missiles. (See Figure 3F-2 for an aerial of this area). Valves upstream of the heat exchangers are manual with a normal lineup that maintains separation of the redundant systems. Downstream valves are normally open, air operated and fail open. Thus damage to the air operators will not result in valve closure. Accordingly, tornadic debris protection of the ICW valves in the CCW heat exchanger area is not necessary.

4.3.2 Component Cooling Water System

The CCW heat exchanger area is not provided with physical missile protection in toto. The CCW heat exchangers and the CCW pumps are potentially exposed to horizontal missiles, thus separation and redundancy are utilized to achieve tornado missile protection. (See Figures 3F-1 and 3F-2 for aerials of this area and section 6.3 of this Appendix for justification). Except for connections to the heat exchangers and pumps, physical protection of valves and piping for horizontal missiles is provided. Figure 9.2-2 provides the CCW system P&ID.

The CCW heat exchangers and piping thereto have adequate capability to accommodate tornadic debris. For example, consider the heat exchangers. They are constructed of steel nominally 0.5 inches thick. The capability of these heat exchangers to accommodate the design basis type missiles is provided by Table 4.

The CCW area is afforded appreciable tornado missile protection by existing plant structures. Immediately to the west lies the Fuel Handling Building and the Reactor Building, to the south the Refueling Water Storage Tank and the Diesel Generator Building. Thus, the CCW area is only vulnerable from southeast to northwest, and to high trajectory missiles from remaining directions.

There are three CCW headers, two essential headers (A and B) and a nonessential header. The nonessential supply header (designated N) which is connected to both essential headers during normal operation, is automatically isolated from both the closure of valves I-HCV-14-8A and 8B and I-HCV-14-9 and 10 on a Safety Injection Actuation signal (SIAS). The valves connecting these headers at the suction side of the pumps and downstream side of the heat exchangers are air operated, fail close. Thus, tornadic debris damage to the air operators causing loss of air would result in these valves (I-HCV-14-8A and 8B, I-HCV-14-9 and 10) closing, i.e., they assume the safe position. Furthermore, the loss of any one of these valves will not prevent the isolation of the essential headers from each other.

Motor operated valves I-MV-14-1 and I-MV-14-2 align CCW pump discharge flow with CCW heat exchangers 1A and 1B. Any failure mode of the operators is acceptable; (i) if both valves were closed, pump 1A is aligned to heat exchanger 1A and pump 1B to heat exchanger 1B; (ii) if one fails open and the other closed, two pumps are available to supply the remaining heat exchanger and one pump is available to supply the remaining heat exchanger; and (iii) if both valves fail open, the 1A and 1B heat exchangers are headered together with supply from three pumps available to both heat exchangers. It should be noted that (i) manual valves I-SB-14166 and I-SB-14156 allow isolation of heat exchanger 1A; (ii) manual valves I-SB-14177 and I-B-14160 allow isolation of heat exchanger 1B; and (iii) after isolating a heat exchanger, manual valves I-SB-14178, I-SB-14439, I-SB-14169 and I-SB-14167 allow connecting heat exchanger 1A with the B header and conversely 1B with the A header.

In a similar manner, motor operated valves I-MV-14-3 and I-MV-14-4 align the suction of the three CCW pumps to the A and B headers. A failure mode and effect analysis of these valves indicates that all possible failure modes are acceptable.

In light of the acceptability of the failure modes of the air and motor operated valves, and the flexibility afforded by the manual valves, additional protection of valve operators in the CCW heat exchanger area is not appropriate, i.e., with regard to tornado debris the design is adequate.

As with the ICW motors, the tornado resistance capability of the CCW system was found to be enhanced by providing localized protection for the motor and the pump's associated small bore piping, i.e., the motor-pump assembly. The enclosure, which is described in Section 6.15 of this Appendix, is compatible with motor ventilation requirements, maintenance requirements, and local structures and components.

4.3.3 Diesel Generator Building Openings

The relationship of the Diesel Generator Building with respect to other plant structures is provided by Figure 3F-1. The building is divided internally by a wall running east-west thereby forming two separate enclosures. The compartment to the north houses diesel generator set 1A and to the south 1B. The air intake louvers for set 1A are on the north face of the building. The air intake for 1B on the south face, and radiator, exhaust and access openings are on the east and west faces of the buildings.

The Diesel Generator Building walls and roof are reinforced concrete 2 feet and 1 1/2 feet thick, respectively. Radiator, access intake and exhaust openings are protected by missile shields whose designs accommodate the access requirements of the road immediately to the east and west of the building, minimize recirculation of exhaust flow, and are compatible with ventilation requirements.

The radiator protection is reinforced concrete that directs the radiator exhaust vertically. The diesel engine exhaust is directed into the plenum formed by the radiator protective structure. Both engine and radiator exhaust flows are released to atmosphere at about roof level (elevation +49'). Drainage at the bottom of the plenum is provided.

The protective structure over the air intakes is of concrete and is so constructed as to force intake air to change direction prior to entering the building. Inlet louvers and steel louver protection are provided in this structure.

A removable steel plate is fitted to each maintenance opening on the east face of the building. Personnel access openings are capable of accommodating tornado winds, thus adequate tornadic debris protection is provided.

4.3.4 D. 0. Fuel Storage and Transfer System

The D. 0. tank area is shown in Figure 3F-1, and Figure 9.5-2 provides the system P & ID. A low concrete wall surrounds the D. 0. tanks, but does not enclose the D. 0. transfer pumps (lA & 1B). Transfer pump 1A is located just outboard of the north face of the low concrete wall and 1B just outboard of the south face. All system valves are manual or protected; thus vulnerability of valve operators is not a consideration.

The D. 0. storage tanks are redundant and sufficiently separated to provide an acceptable level of tornado resistance capability. They are constructed of 0.25" thick steel plate. Based on methodology provided by reference (9), the penetration resistance of these tanks has been determined and is provided in table 2 for the design basis type missile. Each tank has considerable inherent capability to resist penetration of these missiles. Accordingly, tornadic debris capability is considered adequate.

Review of the D. 0. tanks inherent capability indicated that the tornado resistance capability of both Unit 1 and Unit 2 would be enhanced by cross connecting the storage tanks for both units. (See section 6.7 of this appendix and section 9.5.4.2).

The motors for D. 0. transfer pumps 1A and 1B and other vital pump equipment are widely separated. Nonetheless, enhancement of tornadic debris capability results from providing local protection for this equipment. This additional capability is described in Section 6.8 and 6.15.2 of this Appendix.

4.3.5 Auxiliary Feedwater System

A schematic of the Auxiliary Feedwater System is provided by Figure 10.5-2. Two areas of the facility contain outdoor components of this system. The condensate storage tank is located just west of the Turbine-Generator Building (see Figure 3F-3), and the auxiliary feedwater pumps (A, B, and C) are located under the main steam trestle. (See Figures 3F-4 and 3F-5 for an aerial showing the location of the steam trestle.)

The condensate storage tank is protected from horizontal missiles by a 2 foot thick reinforced concrete wall. The top of this tank is not protected from vertical missiles. Since the top of the tank is more than 30 feet above grade, it is not vulnerable to impact from an automobile or utility pole. It is potentially vulnerable- to other high trajectory design basis type missiles. However, only those missiles striking within a small solid angle with the vertical (cone like distribution) are potentially of concern. In addition it must be noted that due to the tank's 0.25" thick steel skin, velocities must be greater than the values specified in Table 2 for the D. 0. storage tank to cause penetration of the tank's top. Penetration of the tank's top alone is inconsequential. The missile must have sufficient energy to penetrate the 0.25" steel top, overcome the depth of water therein, and penetrate another 0.25" steel plate. Thus, due to the limited strike angle and the relatively high threshold energy, the probability of penetration of the top of this tank resulting in an unacceptable loss of water by design basis type missiles is sufficiently small. The tank top's inherent ability to accommodate tornadic debris is adequate.

The acceptably low probability of strike from certain vertical design basis type missiles and the adequate resistance of the tank top to penetration by tornadic debris notwithstanding, the impact of vertical missiles has been analyzed. The study indicated that the worst impact case for a missile striking both the tank top and tank sidewall occurred at an angle with the vertical of 45 degrees. For this case the velocities with no credit for water required to penetrate the tank top and sidewall are greater than for a vertical missile that is capable of piercing the tank top and bottom. Thus, the vertical missile is considered limiting.

The analysis of the condensate storage tank to accommodate a vertical missile assumed 14 feet of water within the tank (about 125,000 gallons). Wooden missiles that penetrate the tank top are stopped by the water. Small steel missiles such as the 1 inch steel rod are slowed down by the water. On the other hand, long design basis type missiles such as the 15 foot pipe sections are relatively unaffected by the 14 feet of water. The results of this analysis is provided in Table 5, which indicates that the condensate storage tank has substantial capability to accommodate design basis vertical missiles of very low strike probability. The design in this regard has been reaffirmed by this analysis. It is adequate.

The auxiliary feedwater pump area is located below the main steam and feedwater pipe trestle. This massive structure provides significant missile protection against vertical missiles. In addition the pump area is afforded considerable protection by plant structures. It is tucked under the steam trestle between the Reactor and Turbine Buildings. The Reactor Auxiliary Building lies to the south and the service building to the north. The electric pumps (A and B) are located in a low-walled off area under the southern trestle section and the steam pump (C) in a similar area under the northern trestle section. The separation of redundant pumps in conjunction with the excellent protection afforded by plant structures provides an acceptable level of protection against tornadic debris

The valves in the Auxiliary Feedwater System are motor operated. Thus, tornadic debris damage to an operator could conceivably result in a valve assuming other than the desired position and/or the inability to change valve position. Two valves so closed could temporally affect the availability of one redundant system. Because of the protection afforded the motor operators by their location in the steam and feedwater trestle area and the fact that two valves must fail to affect the availability of one redundant system, adequate tornadic debris protection is provided for the system's valve operators.

4.3.6 Main Steam and Feedwater System

The main steam and feedwater piping from the containment to the isolation valves is located within and supported by the steam and feedwater trestle structure which consists of horizontal steel beams and columns. The safety related components include the piping, steam line relief valves and containment isolation valves. The steel beams of the trestle structure protect the piping and valves from horizontal tornado missiles. The steel is generally 1.25 inches thick and no less than 0.75 inches thick. The main steam and feedwater piping is exposed to potential missile impact from the vertical direction. The exposed portions of the piping and valve have a carbon steel wall thickness at least 1.25 inches thick. The inherent capability of these components to withstand missile penetration has been evaluated using methods given in References 9 and 10. The missile velocities which the components are capable of withstanding without penetration are given in Table 3. Protection against tornadic debris is adequate. See Appendix sections 6.11 and 6.12 for further justification.

4.3.7 Control Room Air Conditioning System

Figure 9.4-3 provides the control diagrams for the Control Room A/C System.
The outdoor segments of chiller units HVA-3A, HVA-3B, and HVA-3C are located on the roof (elevation 62') of the Reactor Auxiliary Building just east of the control room area structure (see Figure 3F-1). The chiller design is such that missile protective structures could adversely affect chiller performance. Accordingly loss of chiller function was Assumed for this evaluation. Section 9.4.1.2 discusses the loss of all three chiller units. With 93°F outside ambient air temperatures, a maximum control room air temperature of 125°F is reached 54 minutes after loss of all three chiller units. Since equipment within the control room is designed for 130°F, and habitability is not precluded, loss of all three chiller units is acceptable. It is also noteworthy that the plant can be brought to and maintained in the hot safe shutdown condition from outside the control room. Thus protection for tornadic debris is not required.

5.0 SUMMARY AND CONCLUSIONS

The evaluation of the St . Lucie Unit 1 design as licensed for construction under CPPR-74 (issued July 1, 1970) for its inherent capability to sustain tornado strikes has indicated that:

a) Based on meteorological considerations, supported by 85 years of historic data, the tornadic risk associated with the intense design basis type tornado is considerably less in peninsular Florida than in areas to the north and west.

b) The use of outdoor separated redundant components results in an acceptably low probability for loss of redundant components concurrent with a tornado strike at the site.

In addition, the review indicated that enhancement of the facility's tornado resistance capability can be achieved by providing additional protection from tornadic debris, i.e., a spectrum of light miscellaneous objects. Specifically, this involves the following active components and structures:

a) ICW pumps

b) CCW pumps

c) Diesel Generator Building openings

d) Diesel oil transfer pumps

e) Interconnecting the Unit 1 and Unit 2 D. 0. storage tanks

f) Auxiliary Feed Pumps and Interconnecting the Unit 1 and Unit 2 condensate storage tanks.

Protection for tornadic debris in these areas was provided commensurate with the Commission's guidance on backfit as provided by 10 CFR 50.109. The design of the protection satisfies the following criteria:

a) The protection shall not provide an additional source of tornadic missiles, i.e., it will accommodate winds associated with the Staff's 360 mph tornado.

b) The protection will accommodate a seismic event (DBE) without loss of function, or its failure will not adversely affect a safety related component or structure.

c) The protection will be compatible with existing plant structures, maintenance requirements, and component operability requirements.

d) The protection will be compatible with commercial operation of the facility, i.e., implementation of these protective features shall have a minimal effect on the availability of this facility.

TABLE 1

ENHANCED PROTECTION LEVEL CAPABILITY

Local Shielding Penetration Velocity (fps)[2]

Missile	200 mph Tornado (1) Velocity (fps) Max	Avg	1' Reinforced Concrete (3)	1' Reinforced Concrete with 1/4" Steel Spall Plate	1" Steel
2" x 4" x 10' wooden plank	180	145	293	449	454
4" x 12" x 12' wooden plank	150	100	262	395	257
1" Solid Steel Rod	110	94	162	234	359
6" Sch 40 x 15' Pipe	100	95	180	262	259
12" Sch 40 x 15' Pipe	96	95	204	299	259
Utility Pole	95	94	159	230	191

(1) See reference (7) for a discussion of missile behavior in a 200 mph tornado. The maximum and average velocities (fps) are provided here for comparative purposes.

(2) Reference (8) was used to determine concrete penetration capability and reference (9) for steel.

(3) No spalling

Appendix 3F

TABLE 2

D. O. STORAGE TANK

MISSILE PENETRATION RESISTANCE

Missile	Missile Velocity (fps)
2" x 4" x 10' Plank	161
4" x 12" x 12' Plank	102
1" Steel Rod	136
6" Sch 40 x 15' Pipe	92
12 " Sch 40 x 15' Pipe	92
Utility Pole	68

TABLE 3

MAIN STEAM AND FEEDWATER PIPING

MISSILE PENETRATION RESISTANCE

Missile	Missile Velocity (fps)
2" x 4" x 10' Plank	583
4" x 12" x 12' Plank	362
1" Steel Rod	479
6" Sch 40 x 15' Pipe	333
12" Sch 40 x 15' Pipe	336
Utility Pole	248

Appendix 3F

TABLE 4

CCW HEAT EXCHANGER

MISSILE PENETRATION RESISTANCE

Missile	Missile Velocity (fps)
2" x 4" x 10' Plank	265
4" x 12" x 12' Plank	171
1" Steel Rod	220
6" Sch 40 x 15' Pipe	154
12" Sch 40 x 15' Pipe	154
Utility Pole	114

TABLE 5

CONDENSATE STORAGE TANK (1)

MISSILE PENETRATION RESISTANCE

Missile	Missile Velocity (fps)
2" x 4" x 10' Plank	336 (2)
4" x 12" x 12' Plank	336
1" Steel Rod	198
6" Sch 40 x 15' Pipe	130
12" Sch 40 x 15' Pipe	130

(1) Assumes 14 feet of water in tank.
(2) Wooden missiles are stopped by water in tank (see reference 11).

6.0 ADDITIONAL INFORMATION REQUIRED BY NRC STAFF

In Supplement No. 1 to the St. Lucie Unit 1 Safety Evaluation Report, NRC Staff identified items of additional information required for completion of their evaluation of the tornado missile protection capability of the plant and documentation of additional requirements that will be implemented pursuant to its requirements.

The information is provided in the following sections in response to the items which are listed in Table 3-3 on page 3-9 of the SER Supplement No. 1.

6.1 ICW PUMPS (SER Table 3-3 Item 1)

The local missile protection provided for the ICW pumps consists of a 1" thick steel housing over each pump as shown in Figure 3F-6.

Openings are provided at the bottom of the missile protection housing just above the intake structure deck level from elevation 16'-6" to 19'-1". These openings are necessary to ensure proper ventilation air flow for the ICW pump motors. The openings expose only the concrete pump foundations which extend up to elevation 19'-1". The pump casing itself is not exposed adjacent to the ventilation opening.

6.2 ICW PIPES (SER Table 3-3 Item 3)

The ICW pump discharge piping leaves the pump and turns downward immediately. It runs under the intake structure deck, in pipe tunnels and underground until it reaches the heat exchanger areas. The piping is 0.375 inches thick and has the capability to resist large missile penetration as shown in Table 6.

The exposed piping at the intake structure is separated by 28 feet with two ICW pumps and their missile barriers interposed between the redundant piping.

The ICW pump discharge pipes can be isolated by means of protected discharge line cross connect valves located in a valve pit below intake structure deck level. Adequate flow can be maintained through either redundant loop by one of the three ICW pumps.

6.3 COMPONENT COOLING WATER HEAT EXCHANGER (SER Table 3-3 Item 9.4)

Pursuant to the NRC Staff's request, the capability of the CCW heat exchanger to resist tornado missile impactive loadings without unacceptable damage and to maintain function following impact has been evaluated with respect to overall impactive effects, response of connected piping, degree of local deformation and capability to operate with internal damage. The following sections describe the analysis performed, the results obtained and conclusions reached.

It must be noted that the capability to accommodate the large design basis missiles such as the automobile and utility pole is not a design basis for this facility. The discussion supra demonstrates that separation provides an acceptable level of protection. It is fortuitous that these heat exchangers have considerable inherent capability to accommodate these missiles. This is demonstrated in the paragraphs that follow.

6.3.1 OVERALL IMPACTIVE LOADING

To assess the magnitude of loading to which the heat exchanger and its supports would be subjected as a result of tornado missile impact an analysis was performed in which the spectrum of missiles evaluated for penetration effects were postulated to impact the heat exchanger at the maximum acceptable penetration velocities as given in Table 4. In addition, the impact of a 4000 lb automobile travelling at 73 fps was evaluated. The velocity of 73 fps was chosen for the automobile since this velocity was found by NRC to be an acceptable design value for this missile for a recent plant (Reference 13). Detailed missile studies and tornado analyses have been performed for St. Lucie. These establish the propriety of selecting various missile velocities. The value of 73 fps is assumed here for analysis of capability only.

The equivalent static forces resulting from impact of the various missiles at the velocities assumed were calculated using Reference 12, conservatively assuming no penetration (the velocities are those that would analytically penetrate) and using the natural period of vibration of the heat exchanger (0.088 sec). A ductility ratio (μ) of 10 was used although as discussed in Section 6.15 a value of μ=20 could be justifiably used for the materials involved from a purely structural point of view.

The resulting forces and accelerations are shown on Table 9 and compared with the equivalent static horizontal seismic force and acceleration for which the heat exchanger were designed. As indicated, the missile impactive loads are less than the DBE seismic design load for which the heat exchanger and its supports must have been designed to maintain functional integrity. The results strongly suggest that the impactive loading effects of the missiles will not be the limiting damage mechanism. The large impactive loadings associated with the large missiles, particularly the automobile, indicate further analysis is warranted. However, it must be noted that these heat exchangers accommodate these types of missiles thru separation and redundancy. The additional studies are discussed below.

6.3.2 SHELL AND PIPING RESPONSE ANALYSIS

The dynamic response of the heat exchanger shell and interconnected ICW and CCW piping was analyzed using the PLAST computer code to determine resulting stresses and displacements following missile impact. The PLAST code is an elasto-plastic dynamic analysis method of evaluating the response of equivalent piping systems to impactive loading. The program has been approved for use in pipe rupture analysis and is described fully in Section 3.6.4.3.

Figure 3F-11 shows the model used in the analysis. The heat exchanger is modelled as an equivalent horizontal pipe and is represented by nodes 1 through 9 on Figure 3F-11. Each of the elements of the heat exchanger

(9), (18), (26) and (33) are of length equal to the average shell radius and are selected in order to determine the stresses at the junction of the nozzles and shell (nodes 2, 3, 7 and 8). The connected piping is similarly modelled out to the first seismic restraint on each piping run. The physical and material properties assumed for the heat exchanger were obtained from the manufacturer's data.

A total of six loading cases were considered including both the 4000 lb auto travelling at 73 fps and the 1500 lb utility pole travelling at its penetration velocity of 114 fps at each of three locations. The three locations chosen were (refer to Figure 3F-11):

a) Impact at the center of the heat exchanger to its axis (node 5 in the -x direction).

b) Impact at the extreme end of the heat exchanger normal to its axis (node 9 in the -x direction).

c) Impact on the end of the heat exchanger along its axis (node 9 in the z direction).

The cases considered represent the most severe spectrum of modes of impact. The missiles considered are (due to mass and size) the most limiting missiles for impactive effects. Each missile was modelled as an elasto-plastic spring using conservative values for equivalent spring constants.

The 4000 lb auto was considered to have 2 stiffening channels of 1.5 square inch total cross-section and 100 inch in length. Then the elastic spring constant is:

$$K = \frac{AE}{L} = \frac{1.5 \times 30 \times 10^6}{100} = 45 \times 10^4 \ lbs/in$$

Since energy absorption occurs very early in the bumpers and chassis, a yield displacement of 0.001" and a strain hardening slope of the plastic region of the bilinear force deflection curve of 0.04 k were used.

The auto was treated as a uni-axial elasto-plastic strain hardening spring, one end of which was connected to a node point on the heat exchanger. The auto mass was concentrated at the other end of the spring. This mass was given an initial velocity of 73 fps towards the heat exchanger and the system consisting of the piping, heat exchanger and auto was allowed to respond to the resulting impulse.

The maximum kinetic energy imported by the auto to the system occurs just after the auto velocity towards the heat exchanger reaches zero and begins to change duration. Peak stresses in the pipes connecting to the nozzles occur at about this time.

For the utility pole (1.5" in diameter x 35' long weighing 1490 lbs) an equivalent spring was chosen with the following physical properties;

$E = 1.83 \times 10^6$ psi

p = 1830 psi (proportional limit)

Then, the elastic spring constant is:

$$K = \frac{EA}{L} = 6.23 \times 10^5 \ lb/in$$

The slope of the plastic region of the bilinear force-deflection curve is

S = 0.05 K

The wood pole, on impact, releases part of its energy in splinters with 'captured' kinetic energy. This energy loss is modeled as plastic behavior with an effective yield displacement of 0.415".

The pole was treated as a uni-axial elasto-plastic spring with its mass concentrated at one end as in the auto impact model and was given an initial velocity of 114 fps towards the heat exchanger.

The resulting stresses and time of occurrence for the most highly stressed locations are given on Table 10. In all cases the stresses are within yield and less than the ASME III permitted stress of 27,000 psi for emergency loading conditions for Class 2 and 3 components.

The results indicate that the heat exchanger and connected piping are capable of withstanding substantial impactive loading without loss of integrity and that the dynamic effects of impactive loading will not be the limiting damage mechanism when compared to penetration effects.

6.3.3 LOCAL DEFORMATION ANALYSIS

Having evaluated the overall support loading conditions and "far field" dynamic response effects of missile impact and found the results acceptable, the local or "near field" effects were evaluated to determine the degree of local deformation in the shell in the zone of impact. This analysis was performed to determine whether rupture of the shell could occur locally and to determine whether the degree of deformation could result in internal damage to heat exchanger tubing due to shell impingement. It should be noted that the velocities selected are those that analytically predict penetration. The discussion that follows demonstrates that the less rigorous approach utilized in the penetration analysis is conservative.

The case considered involved impact of the 4000 lb automobile at 73 fps normal to the shell at the center of the heat exchanger, midway between the supports. This missile was considered to be the most severe with respect to local deformation and the point of impact was chosen since it is the most critical with respect to tube damage and is the point of impact where greatest local deformation would occur.

The analysis was performed using the ANSYS computer program (Reference 14). Triangular plastic shell elements were used to model the shell and three dimensional plastic isoparametric elements were used to model the automobile. The geometric models are shown on Figures 3F-12, 3F-13 and 3F-14. An initial velocity of 73 fps was given to the auto and it was allowed to impact upon the heat exchanger as shown in Figures 3F-13 and 3F-14. The time dependent analysis included the effects of large deformation and material non-linearities. There were a total of 282 elements and 1083 degrees of freedom included in the model.

Structural damping was assumed to be 3 percent and appropriate factors were applied to the system mass and stiffness materials to form the damping matrix for the entire structure.

The analysis was carried out to a time of 0.01 seconds. At this time the missile velocity has been reduced to 16.5 fps as shown in Figure 3F-15. At this point the velocity gradient is nearly linear and by extrapolation, a zero velocity condition would be expected at about 0.015 second.

The results at 0.01 seconds indicate that the heat exchanger is still sound from a structural standpoint even though deformations have occurred in the zone of impact. Figure 3F-16 shows the radial displacement at various elements in the impact zone vs. time. At 0.01 seconds a maximum displacement of 4 inches has occurred. At 0.01 seconds the displacement curve is approaching an asymptotic value and a total maximum displacement of less than 4.5 inches would be expected. This maximum deflection occurs in the shell directly under the impact zone. This is a local effect which does out away from the impact zone. The peak generalized strain occurs in element 149 which is along the.border of the impact zone. At 0.01 seconds the strain in element 149 is 6.2 percent which compares to an ultimate strain of 22 percent for mild steel. According it is concluded that the shell will not rupture due to impact.

Some yielding also occurs in the area of the supports. In this region the peak generalized strain at 0.01 seconds is 0.94 percent, well below ultimate of 22 percent. All other areas remain elastic with stresses well below yield. Although it has been demonstrated that shell rupture will not occur, the internal tube arrangement is such that impingement of the tube bundle will occur in the impact area. For a 4.5 inch radial displacement, the maximum number of tubes which could be affected is 38 which is approximately 2 percent of the total number of tubes (1950).

The tube damage resulting from shell impingement may in many cases be limited to bending or flattening without rupture due to the ductility of the tubes. Adequate heat transfer capability exists even if all of the 38 tubes affected by the missile strike are completely blocked. The ability of the heat exchanger to perform its heat removal function in the present of tube damage was further investigated and this evaluation is discussed in the following section.

The foregoing analysis also indicates, based on energy absorption considerations, that the heat exchanger can withstand the 4000 lb automobile at a velocity of 30 fps without tube damage.

6.3.4 OPERATION WITH DAMAGED TUBES

The following analysis was performed to demonstrate that it is possible to cooldown the plant with a single, damaged CCW heat exchanger. It is not intended that this analysis be directly incorporated in plant procedures. Post-event operator actions in response to a tornado would be dictated by existing plant conditions and available equipment. Specific recovery actions beyond that provided in the plant Emergency Operating Procedures (EOPs) and the Emergency Plant (i.e., E-Plan), including operation with one damaged CCW heat exchanger, are typically provided by the Technical Support Center Staff.

In the event of missile impact on the heat exchanger which results in the tube damage, the result would be outleakage of water from the shell side (CCW System) to the tube side (ICW System) since there is normally a pressure differential of approximately 35 psi from the CCW side to the ICW side of the heat exchanger.

Depending on the number of tubes ruptured and the availability of CCW makeup sources, the intelligence to identify and assess the magnitude of the leak will be provided by CCW surge tank level and makeup flaw instrumentation. Emergency CCW makeup water is available from the demineralized makeup water system and the city water tanks (1,000,000 gallons capacity) via a normally closed cross-connection from the fire protection system. (See Figure 9.2-2 for CCW System P & ID and Figure 9.2-5 for the makeup water system P & ID). Although these tanks and the fire pumps are not protected against tornado missiles and are not in the final analysis relied upon for demonstrating continued functioning of the CCW system following heat exchanger tube damage, a discussion of their effect on the action to be taken in response to the incident is pertinent. In addition, since the tanks and fire pumps are widely separated from the CCW area on site by over 500 feet (see Figure 1.2-2), there is a high probability that emergency makeup sources will be available following tornado missile impact at the CCW area since the tornado would have to cross the plant in a narrowly defined path in order to cause damage in both areas.

The leakage flow rate out of the CCW system in the event of tube damage would be approximately 50 gpm per tube assuming complete tube severance and double ended flow. The capacity of the emergency makeup line to the surge tank is approximately 600 gpm. Therefore, if the makeup source were available it would be possible to continue to maintain inventory in the CCW system for up to 28 hours with tube damage equivalent to 12 tubes completely ruptured using the water stored onsite. This time could be extended indefinitely since additional water would be available if the 12" supply line from the underground city water supply to the tanks on site remains intact and operable.

In the event of tube damage concurrent with the absence of the normal or emergency makeup, the operator would be alerted to the occurrence of a CCW anomaly by a low surge tank level alarm in the affected redundant subsystem (A or B).

Due to the physical separation between redundant heat exchangers and the inherent capability of the heat exchangers to resist tornadic debris, it would be highly improbable that both heat exchangers would sustain internal damage as a result of the same event. (This is demonstrated quantitatively in Section 3.0 supra.) However, the capability to maintain the plant in a safe condition with internal heat exchanger tube damage has been evaluated assuming that only one heat exchanger (the damaged one) is available following occurrence of the tornado. The procedure that follows is independent of the number of tubes damaged.

If the surge tank low level alarm does not clear within a short period, the operator would, from the control room, isolate the non-essential header and stop the pump in the affected heat exchanger loop. The plant is designed such that those systems and components required to maintain the plant in a safe hot standby condition are not dependent on CCW system operation, e.g., the diesel generators are cooled by missile protected closed loop water-forced air heat exchangers; the auxiliary feedwater pumps are cooled from their own discharge; and control room cooling is provided by air conditioners.

The plant can be maintained at safe hot standby conditions without CCW system operation as long as there is auxiliary feedwater available in the condensate storage tank to provide for decay-heat removal and cooldown to 325F. Assuming the Unit 1 condensate storage tank contains the minimum inventory permitted by plant Technical Specifications, (normally the level is maintained well above the tech spec limit) at the onset of the tornado incident, the operators will have 4 hours to effect the procedures required to place the damaged heat exchanger in service. If the condensate storage tank is full or near full, this time will be as long as 16 hours. The missile protected intertie with the Unit 2 condensate storage tank, when effected (see Section 6.10 herein), will extend these times even further.

There will be substantial time, therefore to investigate the cause of the CCW anomaly, identify heat exchanger tube leakage as the cause and place the system in the requisite operating mode which simply requires balancing of shell and tube side pressures.

As stated previously, the initial consequence of heat exchanger leakage will be a non-clearing surge tank low level alarm which will prompt the operator to isolate the non-essential header and to secure the affected pump. Once the pump is shut down, the level will rise again in the affected CCW loop due to inleakage of intake cooling water until the surge tank level is restored. A high surge tank level will alert the operator to restoration of level. Restart of the CCW pump will again cause the surge tank level to drop. A visual inspection of the heat exchanger shell will confirm that missile impact has occurred. Thus the necessary steps can then be taken to place the heat exchanger in the tornado contingency mode of operation.

Since, with the CCW pump running, the pressure is normally higher on the CCW side of the heat exchanger, it is necessary to locally reposition the CCW pump discharge valve (or alternatively the CCW heat exchanger inlet valve I-SB-14160A or B and the ICW heat exchanger outlet valve TCV-14-4 A or B) to balance the pressure across the heat exchanger tubing. See Figure 9.2-l. The valves involved are located below grade.

The system head and flow characteristics have been evaluated for operation in the above described mode. Figure 3F-17 shows the head in the CCW heat exchanger versus flow for both the shell side (CCW system) and tube side (ICW system) for various conditions of valve throttling in either system. The optimum condition for operation in the emergency mode is with the heat exchanger pressure balanced at about 120 feet head. Under this condition the CCW flow would be approximately 7500 gpm and the ICW flow approximately 7000 gpm. These flows are more than adequate for long term decay heat removal.

To achieve a pressure balance at 120 feet head, the CCW pump discharge valve would be throttled to between 25^0 and 30^0 from the closed position. The ICW valve is normally in throttling service for normal operations. It is placed in a normal operating position. However, it must be noted that it is not necessary to operate with pressure balanced precisely at 120 feet. This is the operating condition at which the surge tank level would be at its normal level. The system can operate satisfactorily over a wide range of CCW system level from a condition with the surge tank full and overflowing to one with the level considerably below the surge tank. Once in the operating mode the inertia of the system is such that rapid changes in CCW level will not occur. Thus the operation can allow the surge tank level to hunt between upper and lower limits.

The initial system operation would involve re-positioning the throttle valves to their approximate positions required to maintain 120 feet head at the heat exchanger. The CCW pump would then be re-started and the surge tank level observed by the operator in the control room. The control room operator would then direct the local valve operator to increase or decrease valve openings depending on whether the surge tank level were rising or falling. This communication can be achieved by direct visual or voice signal. Since there is a wide operating range under which decay heat removal can be assured, such a mode of control is considered entirely adequate for this highly unlikely event.

Once the pressures are balanced or near the balance point, changes in surge tank level will occur slowly. The control room operator can also monitor CCW flow through the shutdown heat exchanger.

The ability of the system pumps and valves to operate in the required mode has been evaluated. The CCW pump characteristic curve is shown on Figure 3F-18. With the CCW head balanced at 120 feet, the pump will be operating very close to the design flow conditions for normal operation. The ICW pump characteristic curve is shown on Figure 3F-19 and with 120 feet at the CCW heat exchanger, the pump flow will be near the post-LOCA design flow conditions. Thus the pumps will not be subjected to extreme or unusual flow conditions.

The CCW pump discharge valve is a 20" butterfly valve and the ICW heat exchanger outlet valve (TCV-14-4 A or B) is a 30" butterfly valve. These valves are provided by the same manufacturer and are suitable for throttling service. The limiting factor for valve throttling application is cavitation which can occur if the pressure drop and flow across the valve are excessive. Reference 15 provides a method for evaluating whether cavitation will occur for a given throttled condition of a butterfly valve. The method involves calculation of a cavitation constant which, if less than 1.0, would indicate serious cavitation. With the 20" CCW valve throttled to 27^0, the required approximate closure position, the calculated cavitation constant is 1.6, well above the point at which serious cavitation can occur. Thus the 20" CCW pump discharge valve is suitable for the throttling service envisioned by this procedure. The manufacturer's valve characteristic data for the 20" CCW valve is given in Table 11.

The ICW heat exchanger outlet valve (TCV-14-4 A or B) is normally in throttling or modulating service. This valve is a temperature control valve which, during normal operation, automatically controls the ICW flow to maintain CCW temperature at a set value. The valve is capable of throttling flow over a wide range of conditions without cavitation. The flow and pressure conditions required for operation in the mode discussed herein are conditions of operation to which the valve will be subjected during normal operation. Preoperational tests have demonstrated that the valve will perform adequately for the required throttling service.

Although the maximum number of tubes which could be damaged as a result of missile impact is about 2 percent of all tubes, the emergency operating procedure described can be effected regardless of the extent of tube damage. As long as pressure balance is achieved, as evidenced by a relatively stable surge tank level, adequate flow for decay heat removal can be maintained through both CCW and ICW systems.

The effects of prolonged operation with salt water on components of the CCW system has been evaluated and it was concluded that, the safety related cooling function of the system will not be impaired for the tornado strike recovery period.

It is concluded that the CCW system may be operated in the manner described to accommodate shell to tube side leakage resulting from internal damage.

6.3.5 GENERAL CONCLUSIONS

The primary means of protection against unacceptable damage to the CCW heat exchanger function due to impact of large missiles has been and remains spatial separation of redundant heat exchangers. This by itself reduces the probability of unacceptable damage to the redundant components to an acceptably low value. The evaluation given in Section 4.3.2 of this Appendix demonstrates the considerable inherent capability of these components to resist tornadic borne debris and even medium to large penetrating missiles. Further the foregoing analysis and discussions in this Section demonstrates the inherent capability of the heat exchangers to resist the impactive effects of the medium to large missiles. Beyond that, it has been further demonstrated that the impactive effects of the 4000 lb automobile at 73 fps can be accommodated via emergency procedures. Therefore, it is concluded that the analyses discussed supra demonstrate the acceptability of the in situ CCW heat exchanger design.

6.4 CCW PIPES (SER Table 3-3 Item 5)

The CCW piping is carbon steel 0.375 inches thick and has capability to resist large missile penetration as shown in Table 6.

6.5 DIESEL GENERATOR BUILDING AIR INTAKES (SER Table 3-3 Item 9)

All exposed ventilation and radiation air intake and exhaust openings have been backfitted with reinforced concrete missile shielding as shown on Figures 3F-7A through 3F-7F. The concrete missile barriers are a minimum of 1 foot thick. To enhance the penetration resistance of the barrier in accordance with the NRC Staff's requirement, a 1/4 inch steel plate has been provided on the inner surfaces of all 1 foot thick section of the missile shielding. Table 1 has been revised to show the missile penetration resistance of 1 foot thick concrete shielding with the 1/4 inch spall plate. The values shown are the velocities at which the various missiles will just perforate the concrete as calculated by the modified Petry formula (Reference 8). Since the steel plate will prevent spalling no further concrete thickness allowance is required to prevent spalling in determining maximum acceptable velocities. Whereas, those shown for concrete alone are based on penetration of 1/2 thickness to accommodate spalling.

6.6 DIESEL GENERATOR BUILDING PERSONNEL ACCESS DOORS (SER Table 3-3 Item 13)

The personnel access doors to the diesel generator building have missile penetration resistance equivalent to that of 1 inch thick steel plate.

6.7 DIESEL FUEL OIL TANKS (SER Table 3-3 Item 14)

A missile protected intertie which includes locked closed isolation valves is provided between the Unit 1 and Unit 2 diesel oil storage tanks. The tie line is installed between the discharge header of the Unit 2 transfer pumps and the discharge header of the Unit 1 transfer pumps. The Unit 2 tanks are missile protected and have a combined capacity of 80,000 gallons. See Section 9.5.4.2.

The average load for safe shutdown conditions for both Units 1 and 2 is slightly less than 2,000 kw. The diesel generator fuel consumption rate at this load is 2.4 gpm. Thus Unit 2 tanks are capable of supplying the safe shutdown fuel needs of both units (with one diesel generator per unit operating) for 11.6 days.

In addition, as explained in Section 9.5.4.1, a contract is maintained with a fuel oil supply and/or shipping company for normal supply of diesel fuel oil. This source would be used under storm conditions if available. In the event that the local firms were also affected by the storm other means and sources of fuel oil would be arranged for in accordance with plant procedures and the site emergency plan. It is extremely unlikely that a single tornado could damage all means of land access to the site. However, in the event there were no bridges open which would permit trucking of fuel onto Hutchinson Island, a barge containing an on-board pump and hoses can be chartered from a local fuel oil company to transport diesel fuel from the substantial FPL inventory at Port Everglades to the St. Lucie site in about one day.

6.8 DIESEL FUEL OIL PUMPS (SER Table 3-3 Item 15)

The diesel oil fuel transfer pumps have been provided with 1 foot thick reinforced concrete missile housing as shown on Figure 3F-8. A 1/4" steel plate is provided on the inside surfaces of all 1 foot thick concrete barriers as required by NRC Staff to enhance missile penetration resistance as discussed in Section 6.5 above.

6.9 AUXILIARY FEEDWATER SYSTEM COMPONENTS (SER Table 3-3 Item 16)

In accordance with NRC Staff requirements (see Figure 3F-9), the area containing the auxiliary feedwater pumps and associated equipment under the mainsteam and feedwater trestle has been provided with 1" thick steel plate missile shielding extending from the trestle horizontal support beams to grade level on the north, south and west sides of the trestle.

6.10 CONDENSATE STORAGE TANK (SER Table 3-3 Item 17)

In accordance with NRC Staff requirements, a missile protected intertie which includes locked closed isolation valves has been provided between the Unit 1 auxiliary feedwater pump suction line and the Unit 2 condensate storage tank.

The Unit 2 tank is fully missile protected and has a capacity sufficient to provide the necessary amount of feedwater for safe shutdown of both Units 1 and 2. Unit 1 and Unit 2 requires a minimum of 130,500 and 154,000 gallons respectively, to achieve RCS cooldown to 325°F at which time decay heat removal can be achieved by initiation of the shutdown cooling system.

6.11 MAIN STEAM AND FEEDWATER PIPING TRESTLE (SER Table 3-3 Item 18)

The main steam trestle is a massive steel framed structure consisting of rigid bents composed of 10' deep built-up girders supported by 4' deep x 2' wide built-up columns.

Under impactive missile loading, some local overstressing will be experienced but the maximum impactive load of approximately 148.5 kips associated with a 4000 lb automobile at 73 fps is not controlling with respect to the overall integrity of the structure when compared with the pipe rupture loads of the order of 700 kips considered in the design of the trestle. Therefore the structure will not experience loss of function under impactive loading.

6.12 EXPOSED MAIN STEAM ISOLATION VALVES (SER Table 3-3 Item 19)

Pursuant to an NRC Staff request both main steam line isolation valves have been assumed to remain open in order to assess the maximum steam line break area that can be tolerated without main steam line isolation.

The maximum break area is limited by the rate at which steam can be vented from the secondary system. The steam vent rate is in turn limited by the rate at which heat can be removed from the reactor coolant system. The maximum rate of heat removal from the coolant system depends on the decay heat generation rate and the allowable cooldown rate within which reactor coolant inventory and reactivity margin can be maintained.

Given the decay heat generation rate and maximum allowable cooldown rate, the steam vent rate vs. time can then be determined from reactor trip until cooldown to 300°F at which time heat removal can be controlled by means of the shutdown cooling system. The maximum steam line rupture will be that break area through which the mass flow rate at no point in time exceeds the maximum allowable vent rate. The mass flow rate through the break will depend on the secondary system pressure and will vary with time and size of break. The maximum allowable break areas (based on maximum allowable cooldown rate) is 4.69 in^2 (equivalent to the flow area of 6.5 one inch schedule 80 pipes).

The main steam lines have substantial capability to accommodate missile impact. Branch lines to the main steam lines are schedule 80 and also have substantial capability to accommodate missiles (pipe wall thickness are provided by Table 7). There are no exposed branch lines from the containment to the turbine building. The main steam lines enter the turbine building below the turbine deck and are routed therein to the area below the turbine standard. 0ne inch instrument lines (sets of three lines) are provided at five different and separated locations along the main steam lines within the turbine building. Each instrument line is provided with two isolation valves located close to the main steam lines. Immediately under the high pressure turbine chest there are two steam line drain pots (one on each line), two 1 inch lines (one on each line) and two 1- 1/2 inch lines. These lines are all isolatable from the main steam line by manual valves located close to the main steam line. In addition they are located well within the turbine building.

In light of the preceding it may be concluded that even if both main steam isolation valves are assumed to remain open, the facility has acceptable capability to accommodate small steam line breaks. The most vulnerable branch line is a one inch instrument line. A single missile could only strike (if one neglects the protection afforded by the turbine building) a set of three lines. The facility can readily accommodate complete severance of 6-1/2 instrument lines. The inherent capability of the few larger branch lines to accommodate missiles is appreciable. This in conjunction with their location below the high pressure turbine chest, well within the turbine building, provides an acceptable level of protection for these lines.

6.13 PRIMARY SYSTEM MAKE-UP WATER SUPPLY (SER Table 3-3 Item 20)

To accommodate moderator shrink approximately 24,000 gallons of makeup are required to cooldown from hot standby conditions to the shutdown cooling window. A minimum of about 9000 gallons of concentrated boric acid is available from the boric acid makeup tanks (2 tanks - 9975 gallon capacity each), which are located within the reactor auxiliary building. The remaining makeup is normally obtained from the refueling water tank, which is not tornado protected. Accordingly, an additional source of 15,000 gallons of tornado protected water will be made available. Three acceptable means of providing this water were initially considered, namely,

1. Provide a manual connection to the holdup tanks (4 - 40,000 gallon tanks), which are located within the reactor auxiliary building.

2. Provide a connection to the spent fuel pool via the fuel pool cooling and purification system.

3. Provide a new storage tank within the reactor auxiliary building.

However, according to Section 9.3.4.3.1, a SIT to VCT intertie was provided instead of these options.

Cooldown can be initiated with makeup and reactivity control provided from the boric acid makeup tanks. Additional makeup at maximum allowable cooldown rate would not be required for about an hour after commencement of cooldown. Thus, manual actuation of the additional makeup source is acceptable.

For a description of the installed primary system shrink, make-up system, see Section 9.3.4.3.1.

6.14 PENETRATION CAPABILITY OF PROTECTIVE STRUCTURES (SER Table 3-3 General Item a)

The missile penetration resistance capability of the protective missile shielding is shown on Table 1.

6.15 IMPACTIVE LOADING CAPABILITY OF PROTECTIVE STRUCTURES (SER Table 3-3 General Item b)

6.15.1 STEEL STRUCTURES

The 1" steel structures provided for protection of the CCW and ICW pumps and Auxiliary Feedwater System components are shown on Figures 3F-6, 3F-9 and 3F-10.

The-component cooling water pumps are protected from missile impact by 16' x 9' x 9' high enclosures provided around each pump. The protective enclosure consists of 1" thick steel barrier plate supported on a structural steel frame which is composed of W8 beams and columns and 5" structural tee bracing. Additional stiffness for the structure is provided by the vertical barrier plates, which are welded to the framing members.

The intake cooling water pumps are protected from missile impact by 9' x 9' x 20' high enclosures provided around each pump. Each enclosure is a steel framed box consisting of W8 beams, W14 columns and 5" structural tee bracing. A 1" thick steel barrier plate is welded to the frame and provides additional stiffness against wind or missile loading. The barrier plate is extended beyond the frame in order to protect the expansion bellows of the intake cooling water piping.

The motor driven auxiliary feedwater pumps are protected from missile impact by (1) the main steam trestle (discussed above) and (2) the provision of 1" thick steel barrier plate over the open areas on the north, south, and west faces of the, trestle structure. (The east face is protected by the containment building). The barrier plate on each face is supported by a structural steel truss spanning between the trestle columns. The truss framing consists of W33 chord members and W8 diagonal braces. This area is afforded excellent protection by its physical location (see Section 4.3.5 supra). Accordingly a discussion of missile impingement in this area is considered somewhat academic.

Since these structures were intended primarily to prevent tornado borne debris from causing loss of the particularly vulnerable active components which they house, they were not specifically designed to resist large impactive loading such as would be associated with impact of the larger missiles (utility pole and automobile) at high velocities. The CCW and ICW protective structures rely on separation to insure an acceptable level of protection for such large missiles.

To ensure that the structures themselves do not become a source of damage to the protected components, the structures were designed to withstand the maximum credible tornado wind (360 mph) and design basis earthquake (DBE) loading conditions. From the viewpoint of overall structural stability and member load bearing capacity the 360 mph wind loading is the governing design condition.

In response to the NRC Staff's request, the degree of inherent capability of the design to resist tornado missile impactive loading was investigated using the methods of Reference 12 to establish equivalent static loadings for various missile impact conditions. The analytical methods employed are useful in estimating the minimum capability of the structures analyzed. More rigorous methods would demonstrate greater impactive capability. However, their use is not considered necessary. Separation provides the primary means of protection. The analysis discussed below demonstrates significant inherent impactive capability.

Reference 12 is cited by the NRC Staff as a conservative method of determining equivalent static loads for missile impact in Structural Engineering Branch Document (B) (forwarded as enclosure No. 2 to the AEC letter to FPL of August 29, 1973). Reference 12 provides two methods of arriving at equivalent static loads; one in which it is assumed that penetration occurs with impact and one in which no penetration occurs. Since the case with no penetration yields more conservative (higher) static loads, this method was employed using the following expression for static load:

$$q = \frac{2\pi mv}{T}\sqrt{\frac{1}{2\mu - 1}}$$

In which:

q	=	Equivalent static load, lbs
m	=	Missile mass, slugs
v	=	missile velocity, feet/second
T	=	Natural period of structure, seconds
μ	=	Ductility

A value of T = 0.1 sec was used for the protective structures. This is taken to be a reasonable value since the natural periods of frame and plate structures of the type involved are typically on the order of 0.1 sec. A value of μ = 20 was used as recommended by Reference 11. A range of μ = 10 to 30 is recommended in Reference 12 for moderately ductile structures, and a range of μ = 30 to 100 is recommended for very ductile structures. Based on the materials involved a ductility of 20 is considered conservative.

Using the above methods and assumptions the static impact loads for missiles at penetration velocity for one inch steel plate and the automobile at 73 fps were calculated. The equivalent static loads were considered as a single concentrated load applied on the center of the exposed height of the structure under study. The capability of the ICW, CCW and auxiliary feedwater areas to accommodate these impactive loads are given on Table 8. As shown the structures have considerable inherent capability to resist impactive loading. The maximum tolerable impactive velocities are greater than the penetration velocities for 1 inch steel plate for all missiles except the utility pole. The tolerable impactive velocities for all missiles are greater than the maximum velocities associated with the 200 mph historic Florida tornado.

The values shown on Table 8 for the auxiliary feedwater shield are the maximum tolerable loads and velocities for which the principal truss chord members would remain within allowable stresses. The shield structure would not lose its structural integrity at those loadings. It must be noted that under these impactive missile loadings, though, some minor chord members would be overstressed. However, such local overstressing would not result in a loss of protective function or the generation of a secondary missile.

The simplified static methodology used to evaluate impactive loading provides conservative estimates of the maximum tolerable missile velocities. Should more sophisticated analysis methods be employed which account for the actual dynamic loading conditions and energy absorbed by deformation of the missiles, the values shown in Table 8 would be increased.

In addition to the steel protective structures described and evaluated above, there is some steel plate protective shielding provided for the diesel generation building opening as shown on Figure 3F-7.

The diesel generator building equipment maintenance access openings are protected from missile impact by 1" thick steel plate barriers installed in front of each opening. The barrier plate is stiffened by 5" structural tees and is supported in guide channels provided outside each opening.

Under impactive missile loading, flexural yielding of the barrier plate will occur. However, the channel guides will restrain the plate from becoming a secondary missile, and the remaining energy of impact will be dissipated by the membrane action of the plate. Therefore, impactive loading of the steel protective plate will not result in damage to the components housed within the diesel generator building.

6.15.2 CONCRETE STRUCTURES

Reinforced concrete missile barriers are provided on the diesel generator building openings and the diesel oil transfer pumps as shown on Figures 3F-7 and 3F-8. The concrete missile barriers are a minimum of 1 foot thick. An impactive analysis of these concrete structures has been performed utilizing the spectrum of missiles listed in Table 1 at the velocities which the 1 foot of reinforced concrete is capable of withstanding without spalling. Reference 12 was used to calculate equivalent static loads, which were applied as a single concentrated load on the structure. The load resulting from the utility pole at 159 fps is equivalent to that from a 4000 lb automobile at 60 fps. For these loading conditions, the stresses resulting from the missile impact are within allowable ACI-318-63 limits.

As discussed in Section 6.5, pursuant to the request of NRC, a 1/4" spall plate is provided on all 1 foot thick barriers to enhance their penetration resistance. This plate adds to the already sufficient capability of the 1 foot concrete barriers to withstand the impactive loading.

Appendix 3F

TABLE 6

ICW or CCW PIPING[1]
MISSILE PENETRATION RESISTANCE

Missile	Missile Velocity (fps)
2" x 4" x 10' Plank	218
4" x 12" X 12' Plank	138
1" Steel Rod	180
6" Sch 40 x 15' Pipe	124
12" Sch 40 x 15' Pipe	97
Utility Pole	66

(1) Based on 3/8" pipe thickness with correction made for pipe curvature (30" diameter)

TABLE 7

MAIN STEAM LINE BRANCH PIPING
WALL THICKNESS

Nominal Line Size	Wall Thickness (in)
1"	0.179
1 1/2"	0.200
2 1/2"	0.276
4"	0.337
10"	0.593

Appendix 3F

TABLE 8

IMPACTIVE LOADING CAPABILITY
OF PROTECTIVE STRUCTURE

Missile	Maximum Tolerable Velocity (fps) ICW Pump Housing	CCW Pump Housing	Aux F.W. Area Shield
4" x 12" Plank	>1000	>1000	>1000
1" Steel Rod	>1000	>1000	>1000
6" Sch 40 Pipe	995	712	592
12" Sch 40 Pipe	380	273	226
Utility Pole	189	134	141
4000 lb Automobile	69	50	75

Appendix 3F

TABLE 9

COMPARISON OF EQUIVALENT STATIC LOADINGS ON CCW HEAT EXCHANGER AND ITS SUPPORTS

	Kips	Acceleration[(1)]
DBE Seismic Loading	155.2	0.8g
Missile Impactive Loading[(2)]		
2" x 4" Plank - 265 fps	3.74	0.02g
4" x 12" Plank - 171 fps	17.4	0.09g
1" Steel Rod - 220 fps	0.89	0.005g
6" Sch 40 Pipe - 154 fps	22.4	0.11g
12" Sch 40 Pipe - 154 fps	62.7	0.32g
Utility Pole - 114 fps	87.0	0.45g
4000 lb Auto - 73 fps	148.5	0.76g

Notes:

(1) Based on heat exchanger flooded weight of 194,000 lbs.

(2) Except for the automobile, the velocities are those that would result in penetration (as predicted by conservative analytical techniques). The value of 73 fps for the auto (there is no penetration) was arbitrarily selected such that it complies with the staff's guidance for current facilities.

Appendix 3F
TABLE 10
MAXIMUM STRESSES DUE TO MISSILE IMPACTIVE LOADING

Pole: WT. = 1490.0 lbs, VEL = 114 ft/sec
Car: WT. = 4000.0 lbs, VEL = 73.0 ft/sec

		Pole		Car	
Location and Direction of Impact	Location in Pipe Element	Stress (psi)	Time (sec)	Stress (psi)	Time (sec)
Case I	Node 11	1457.5	0.0549	507.2	0.105
	Node 20	186.9	0.0639	179.6	0.105
At Node 9 In	Node 28	302.8	0.0909	309.4	0.105
+ Z Direction	Node 35	1035.2	0.0819	457.8	0.114
Case II	Node 11	14693.5	0.0693	13105.9	0.078
	Node *20*	10242.5	0.114	1206.5	0.168
At Node 9 in	Node 28	20991.1	0.0872	14104.5	0.096
- X Direction	Node 35	16621.4	0.105	22945.8	0.096
Case III	Node 11	1226.6	0.105	1051.2	0.159
	Node 20	806.6	0.123	549.8	0.15
At Node 5 In	Node 28	2226.9	0.141	1281.3	0.177
- X Direction	Node 35	636.5	0.105	763.2	0.141

30" Pipe (Nodes 11,20):
O.D. = 30.0" R_o 15.0. I.D. 29.25", R_i - 14.625", Wall Thickness = 0.375"
I = 3830.98614 in 4, Z = I/R = 3830.98614/15.0 = 255.399 in^3

24" Pipe (Nodes 28,35)
O.D. = 24.0", R_o = 12.0", I.D.23.25" R_i = 11.625", Wall Thickness = 0.375"
I = 1943.08 in 4, Z = 1943.08/12.0 = 161.923 in^3

Appendix 3F

TABLE 11

CHARACTERISTICS OF CCW SYSTEM BUTTERFLY VALVES

Degrees Open	K (see below)
5	15625
10	3860
15	935
20	337
25	145
30	71.8
35	39.6
40	21.6
45	12.7
50	7.42
55	4.41
60	2.64
65	1.59
70	0.952
75	0.620
80	0.496
85	0.460
90	0.447

Head Loss = $(KV^2)/(2g)$

V = Fluid Velocity (ft/sec)

g = 32.2 ft/sec ^{2}

REFERENCES FOR APPENDIX 3F

1) Thom, H.C.S., 1963, Tornado probabilities, Monthly Weather Review, Volume 91, p. 730-736.

2) Howe, G.M., April 1974, Tornado oath sizes, Applied Meteorology, Volume 13, No. 3, p. 343-347.

3) Wilson, J.W., and Morgan, G.M., October 1971, Long track tornadoes and their significance, Preprints, Seventh Conference on Severe Local Storms, p. 183-186.

4) Fawbush, E.J., and Miller, R.C., April, 1954, The types of air masses in which American tornadoes form, Bulletin of American Meteorological Society, Volume 35, No. 4, p. 145-165.

5) Beebe, R.G., April 1958, Tornado proximity soundings, Bulletin of American Meteorological Society, Volume 39, No. 4, p. 195-201.

6) U.S.A.E.C., May 1974, Technical basis for interim regional tornado criteria, WASH-1300 (UC-11).

7) Appendix 3E to St. Lucie Unit #2 PSAR (docket 50-389), Amendment 21, dated October 1, 1974.

8) Amirikian "Design of Protective Structure Structures," Report NP 37,16, Bureau of Yards and Docks, Dept. of Navy, August 1950.

9) Gwaltney, "Missile Generation in Light Water Cooled Reactors," ORNLNSIC-22, March 1, 1967.

10) DR DeBoisblanc, et. al., "RID Report 1003," "The Effect of Small Equipment Missiles on Integrity of Vital Piping," July 1971.

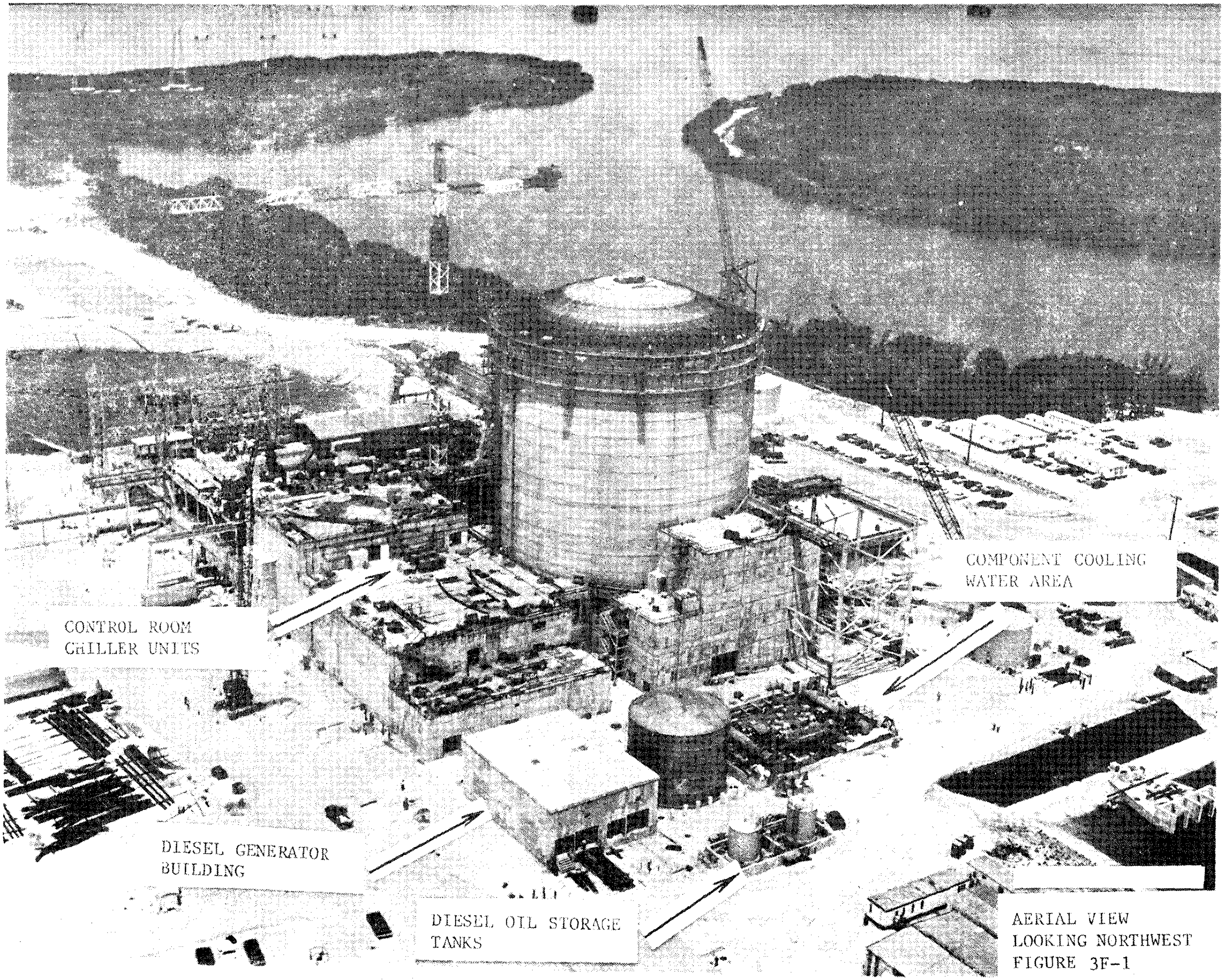
11) "Design of Structures for Missile Impact," Bechtel Power Corporation, BTOP-9 Revision 1, July 1973.

12) RA Williamson and RR Alvy, "Impact Effects of Fragments Striking Structural Elements" NP-6515, Holmes and Narren, 1957.

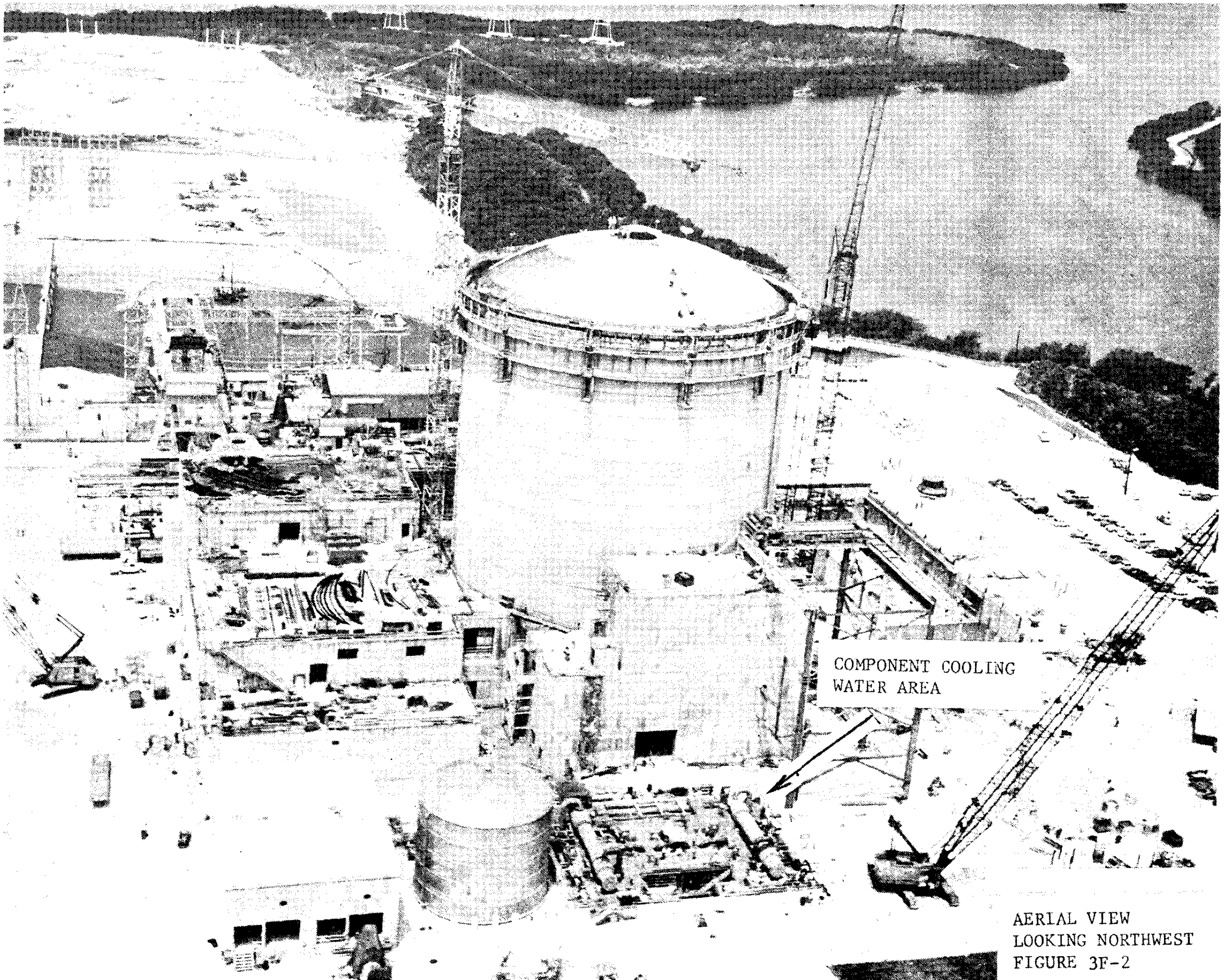
13) Safety Evaluation Report for Greenwood Energy Center (dockets 50-452, 50-453), pages 3-8, July 17, 1974.

14) Swanson Analysis Systems, Inc. Rev. 2, Update 155.

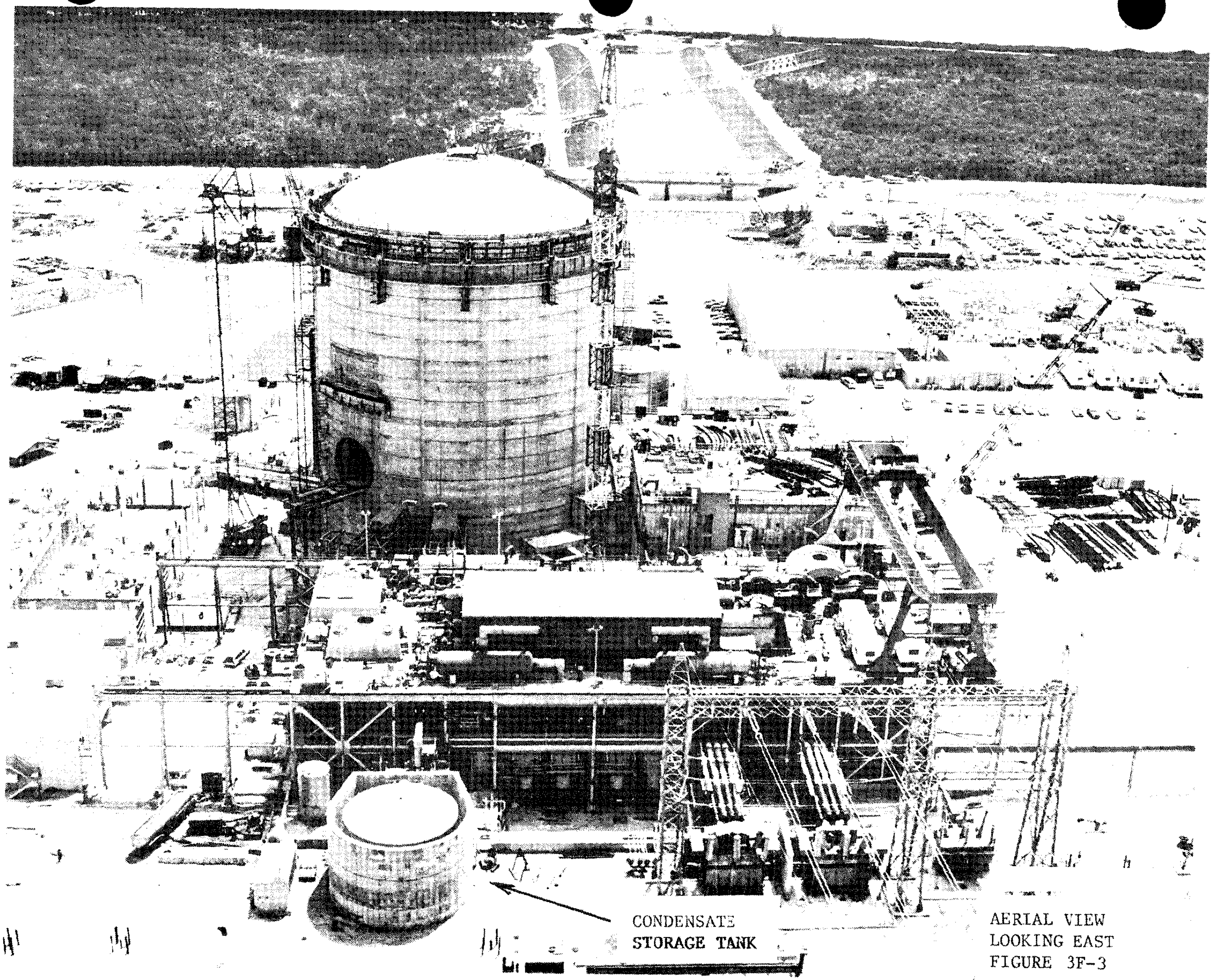
15) <u>Power</u>, July 1974, pp. 42-44.



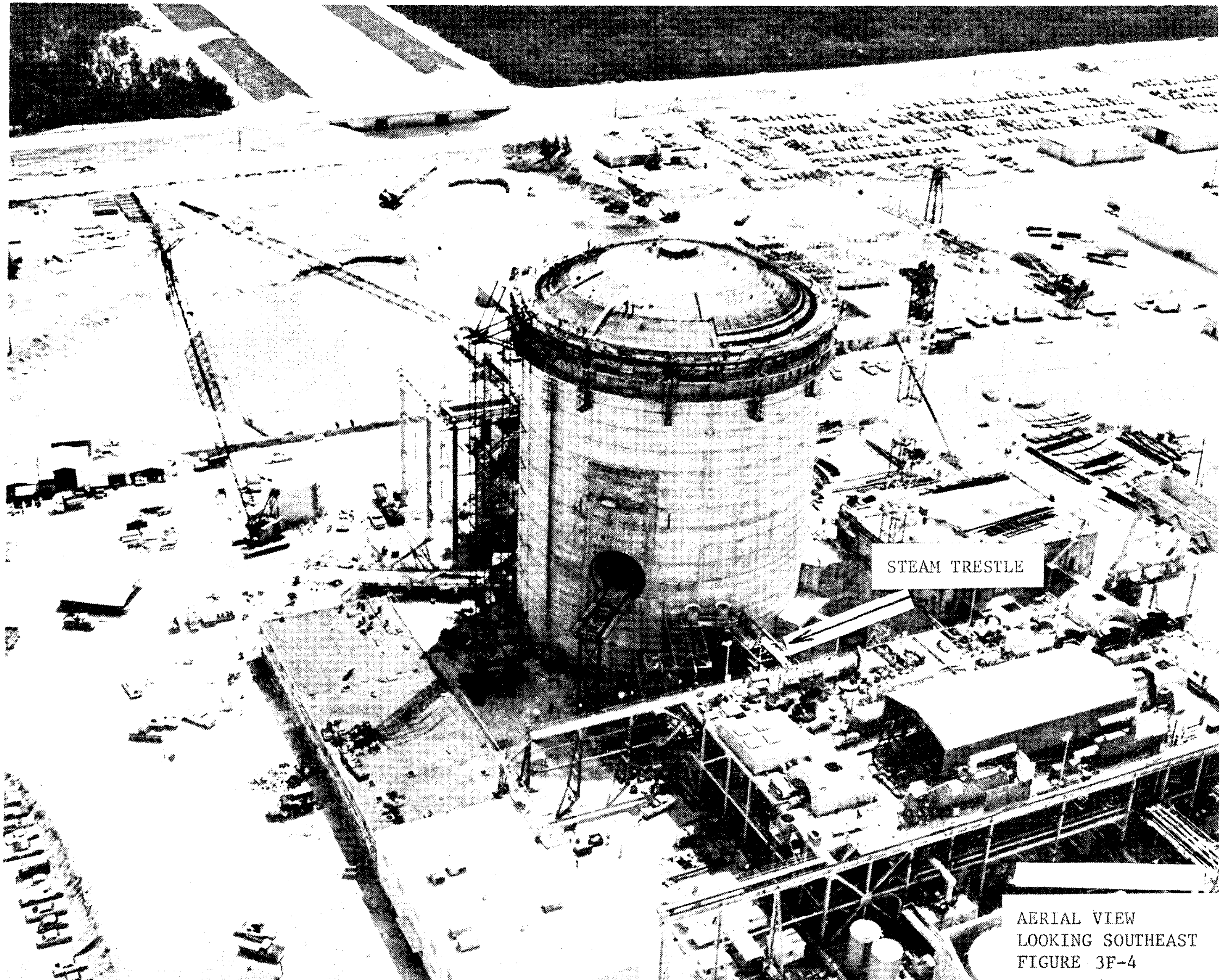
AERIAL VIEW
LOOKING NORTHWEST
FIGURE 3F-1



AERIAL VIEW
LOOKING NORTHWEST
FIGURE 3F-2



AERIAL VIEW
LOOKING EAST
FIGURE 3F-3



AERIAL VIEW
LOOKING SOUTHEAST
FIGURE 3F-4

AERIAL VIEW
LOOKING SOUTH
FIGURE 3F-5

Refer to drawing
8770-G-836 Sheet 1

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MISSILE BARRIER INTAKE PUMP AREA

FIGURE 3F-6

REFER TO DRAWINGS

8770-G-702 SHEETS 1 & 2

FLORIDA POWER AND LIGHT COMPANY **ST. LUCIE PLANT UNIT 1**
TORNADO MISSILE PROTECTION DIESEL BLDG.-MAS **FIGURE 3F-7a**

Refer to drawing
8770-G-703

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

TORNADO MISSILE PROTECTION DIESEL GEN
BLDG-M&R SH 1
FIGURE 3F-7b

REFER TO DRAWING

8770-G-704

Amendment No. 22 (05/07)

FLORIDA POWER AND LIGHT COMPANY ST. LUCIE PLANT UNIT 1
TORNADO MISSILE PROTECTION DIESEL GEN BLDG - REINF SH 2 FIGURE 3F-7C

Refer to drawing
8770-G-705

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

TORNADO MISSILE PROTECTION DIESEL GEN BLDG-REINF SH 3
FIGURE 3F-7d

REFER TO DRAWING

8770-G-706

Amendment No. 22 (05/07)

FLORIDA POWER AND LIGHT COMPANY **ST. LUCIE PLANT UNIT 1**
TORNADO MISSILE PROTECTION **DIESEL GEN BLDG - REINF SH 4** **FIGURE 3F-7E**

REFER TO DRAWING

8770-G-836, SHEET 3

Amendment No. 22 (05/07)

FLORIDA POWER AND LIGHT COMPANY ST. LUCIE PLANT UNIT 1
MISSILE BARRIER DIESEL GENERATOR AREA FIGURE 3F-7F

REFER TO DRAWING

8770-G-707

Amendment No. 22 (05/07)

FLORIDA POWER AND LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

TORNADO MISSILE PROTECTION
DIESEL OIL TRANSFER PUMPS M&R
FIGURE 3F-8

Refer to drawing
8770-G-836 Sheet 4

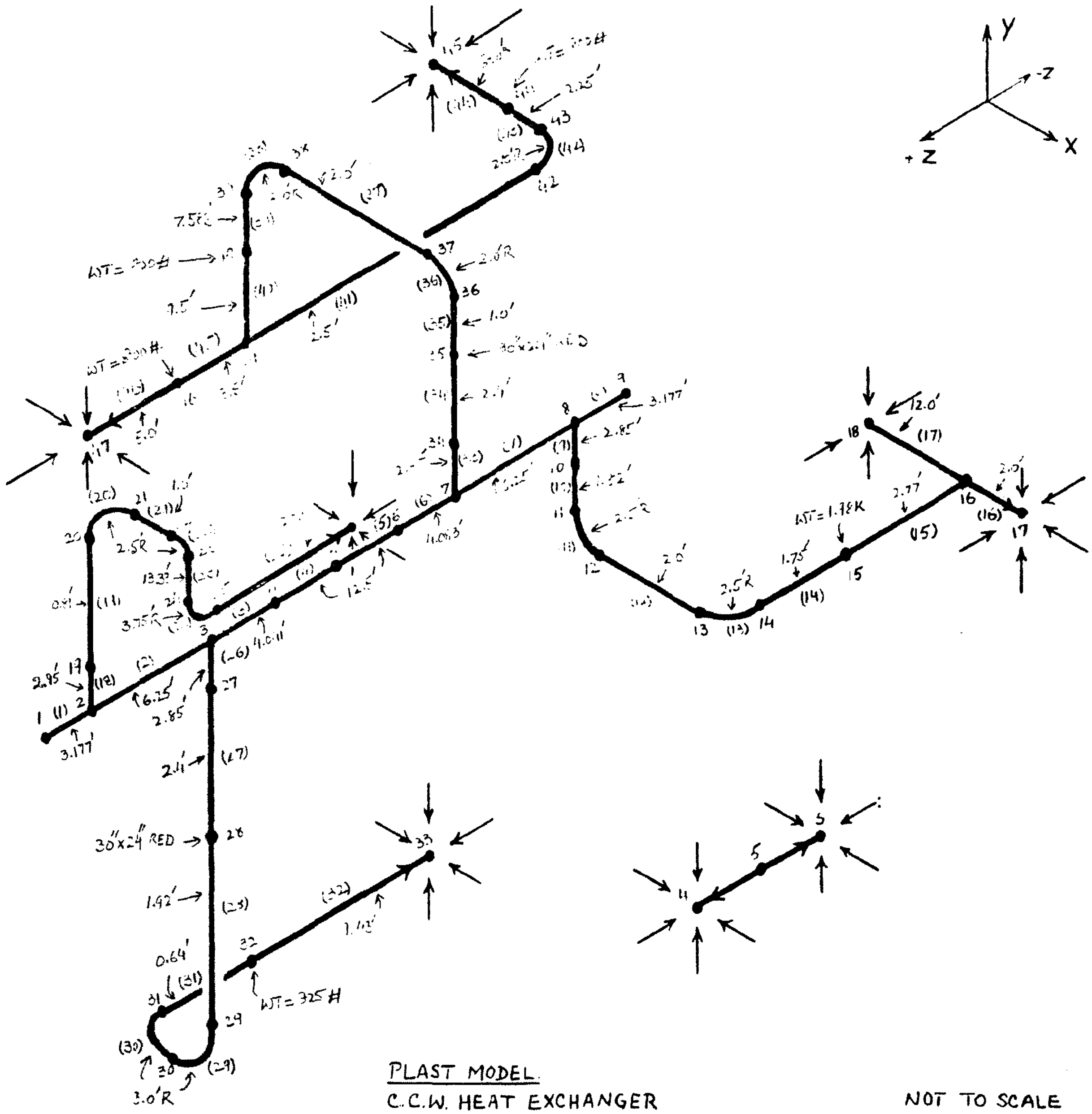
FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT UNIT 1**
MISSILE BARRIER AUX. F.W. PUMPS FIGURE 3F-9

Refer to drawing
8770-G-836 Sheet 2

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

MISSILE BARRIER COMPONENT COOLING AREA

FIGURE 3F-10

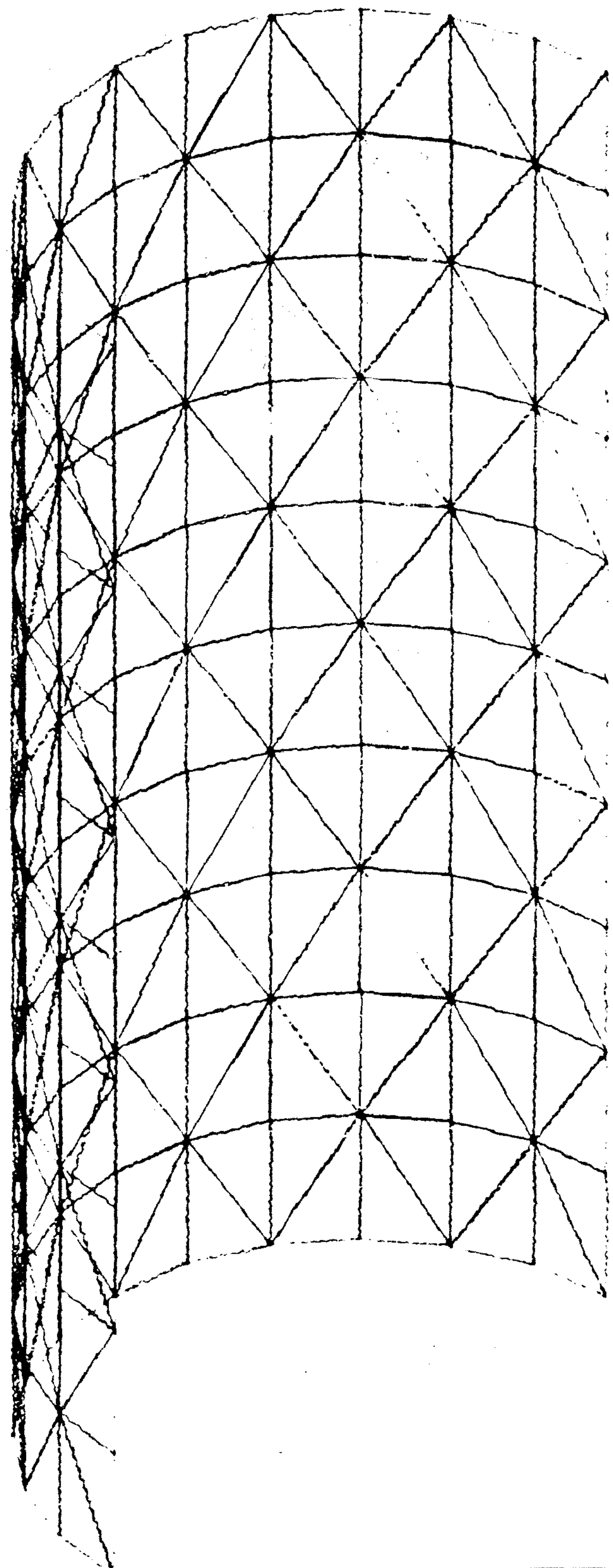


PLAST MODEL

C.C.W. HEAT EXCHANGER AND INLETS & OUTLETS

NOT TO SCALE

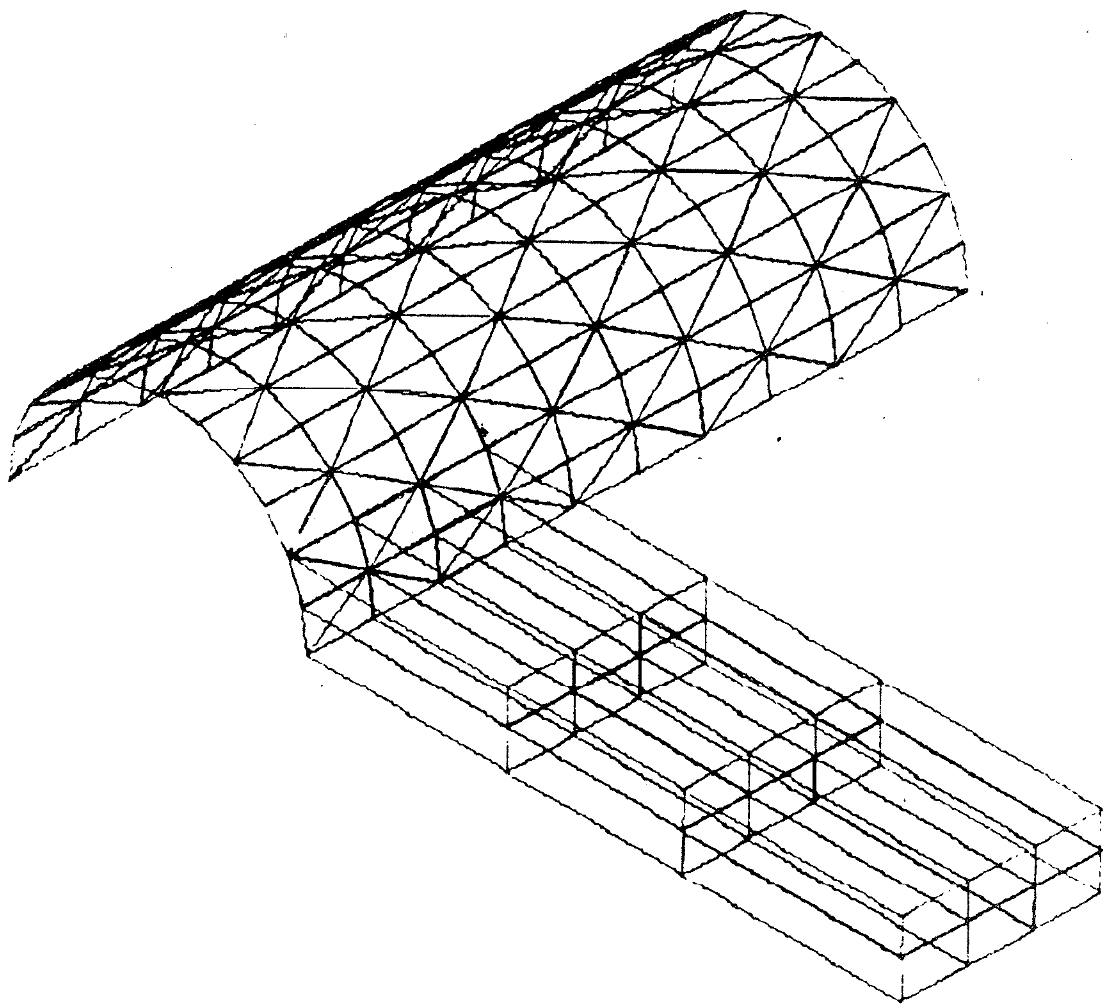
FLORIDA POWER & LIGHT COMPANY ST. LUCIE PLANT UNIT 1
PLAST MODEL OF CCW HEAT EXCHANGER AND CONNECTED PIPING FIGURE 3F-11



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FINITE ELEMENT OF CCW HEAT EXCHANGER SHELL

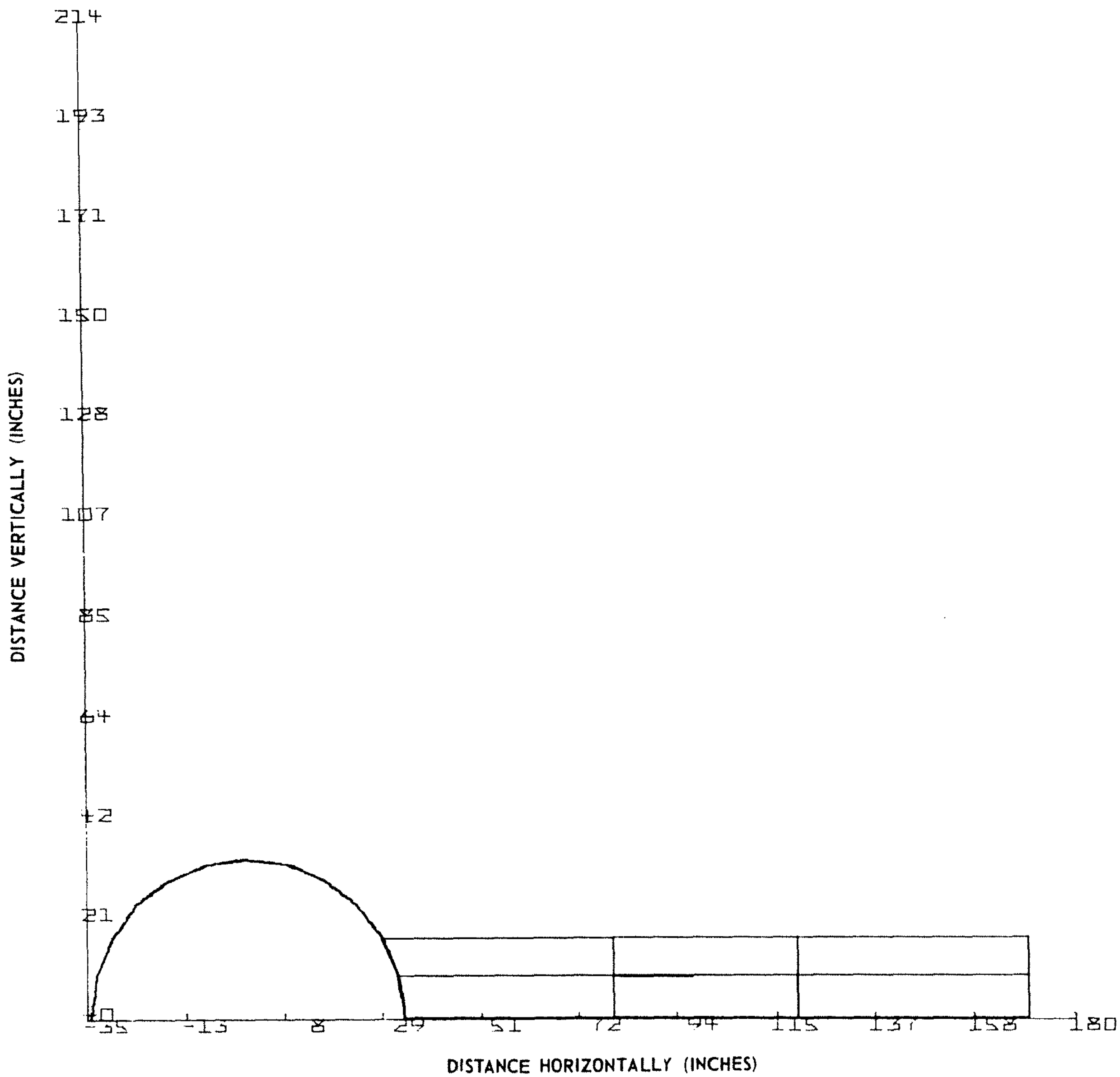
FIGURE 3F-12



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FINITE ELEMENT ISOMETRIC MODEL OF CCW HEAT EXCHANGER AND AUTOMOBILE

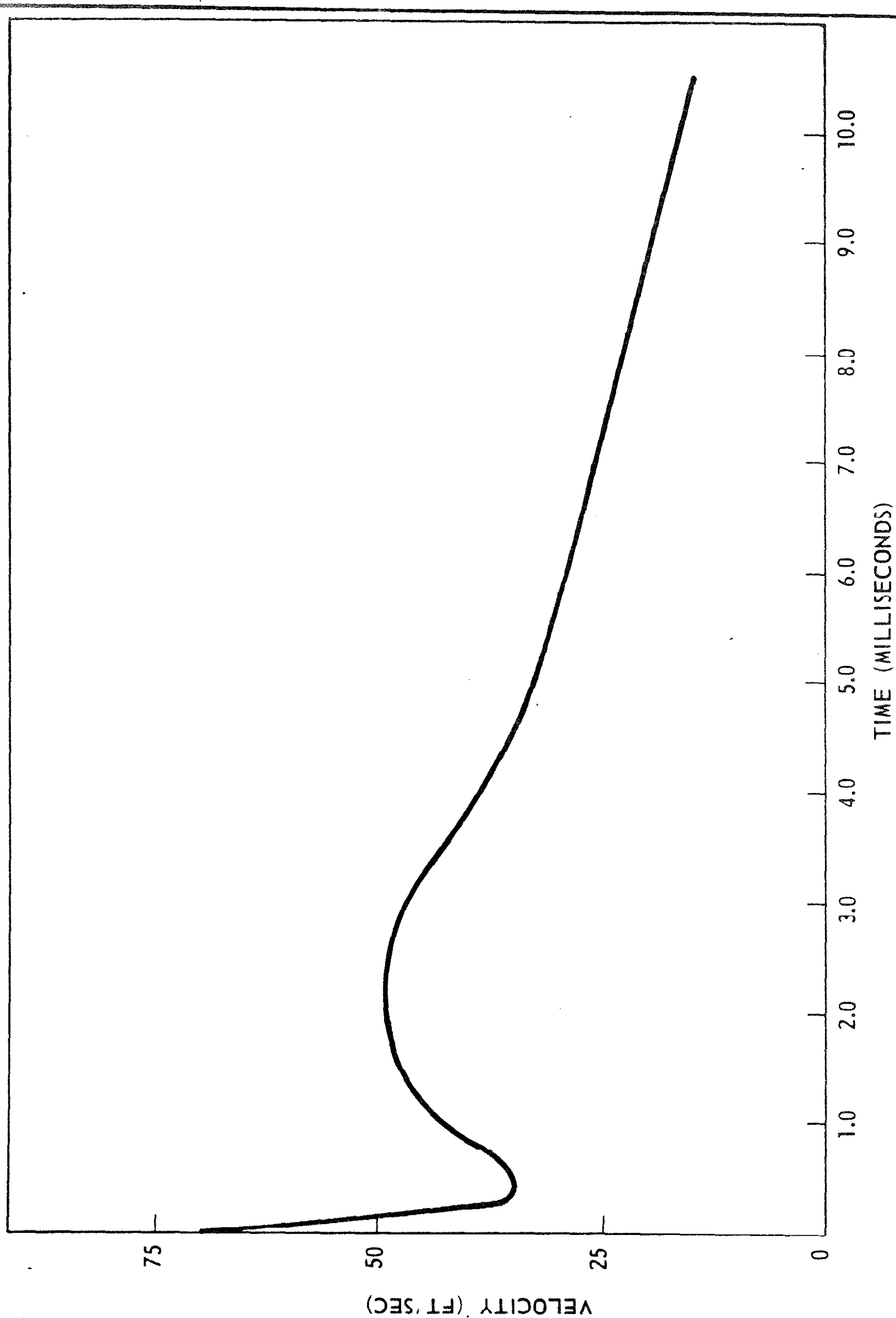
FIGURE 3F-13



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

CCW HEAT EXCHANGER - AUTOMOBILE MODEL VERTICAL PLANE

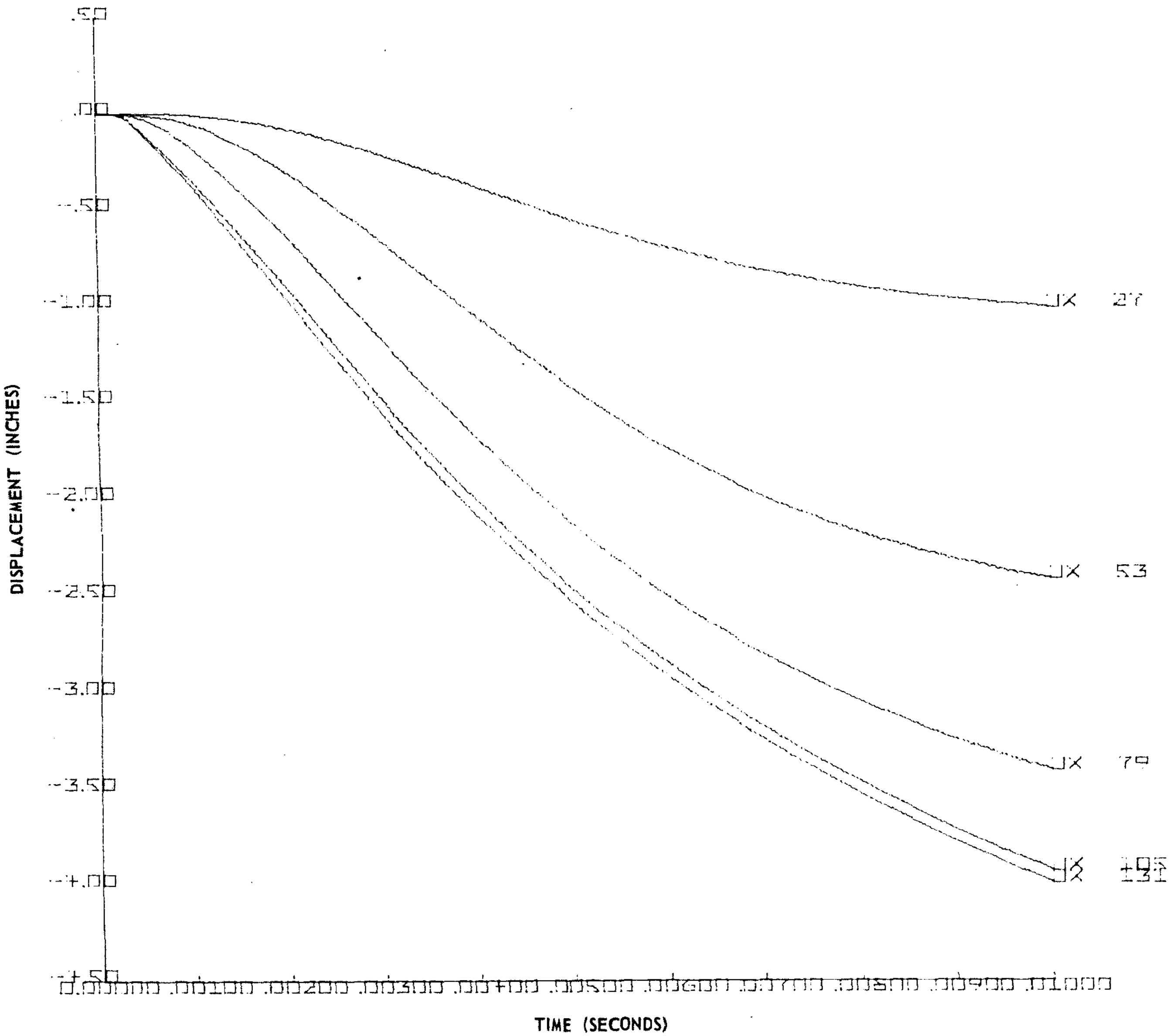
FIGURE 3F-14



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

VELOCITY TIME HISTORY
OF FRONT OF AUTOMOBILE

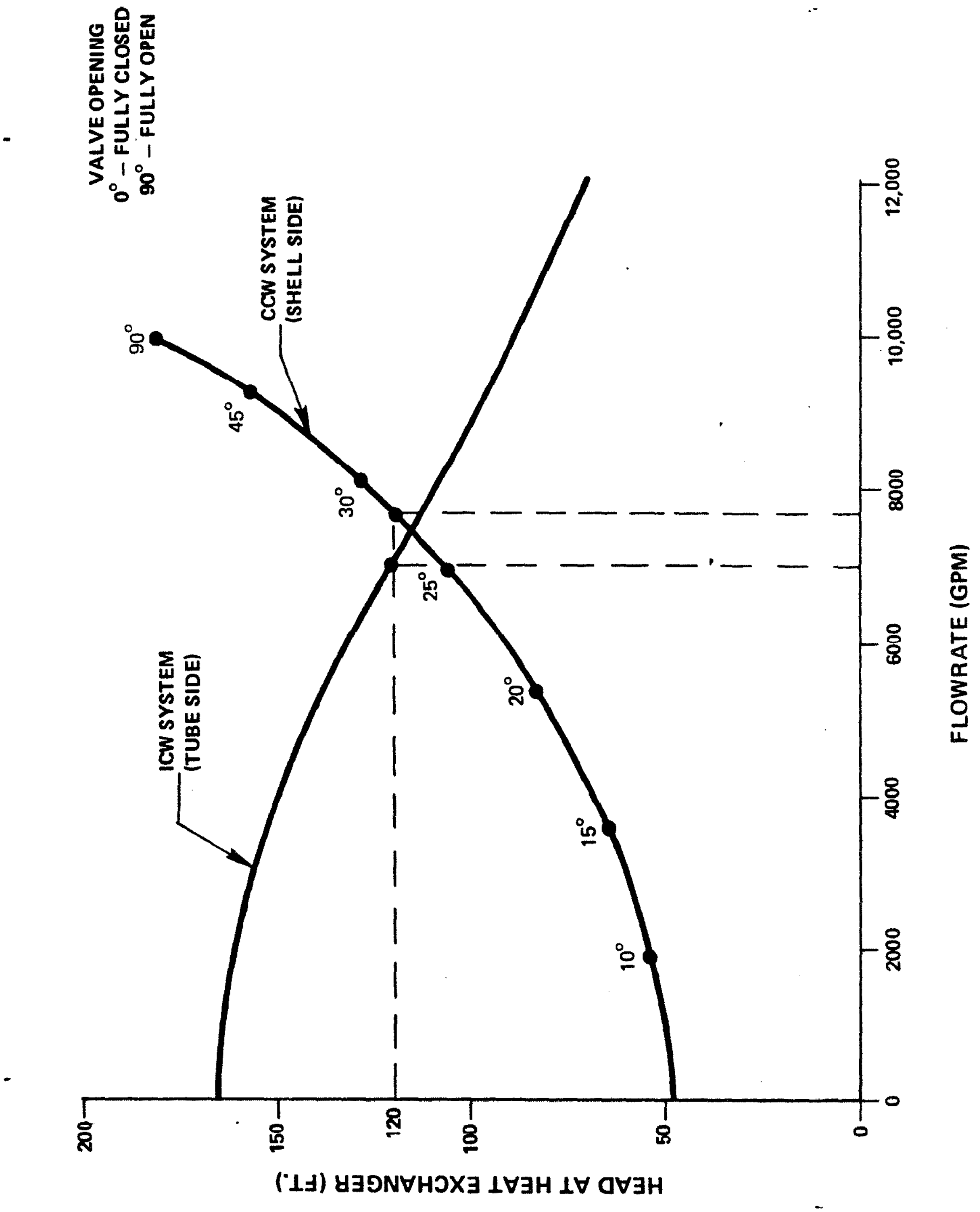
FIGURE 3F-15



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

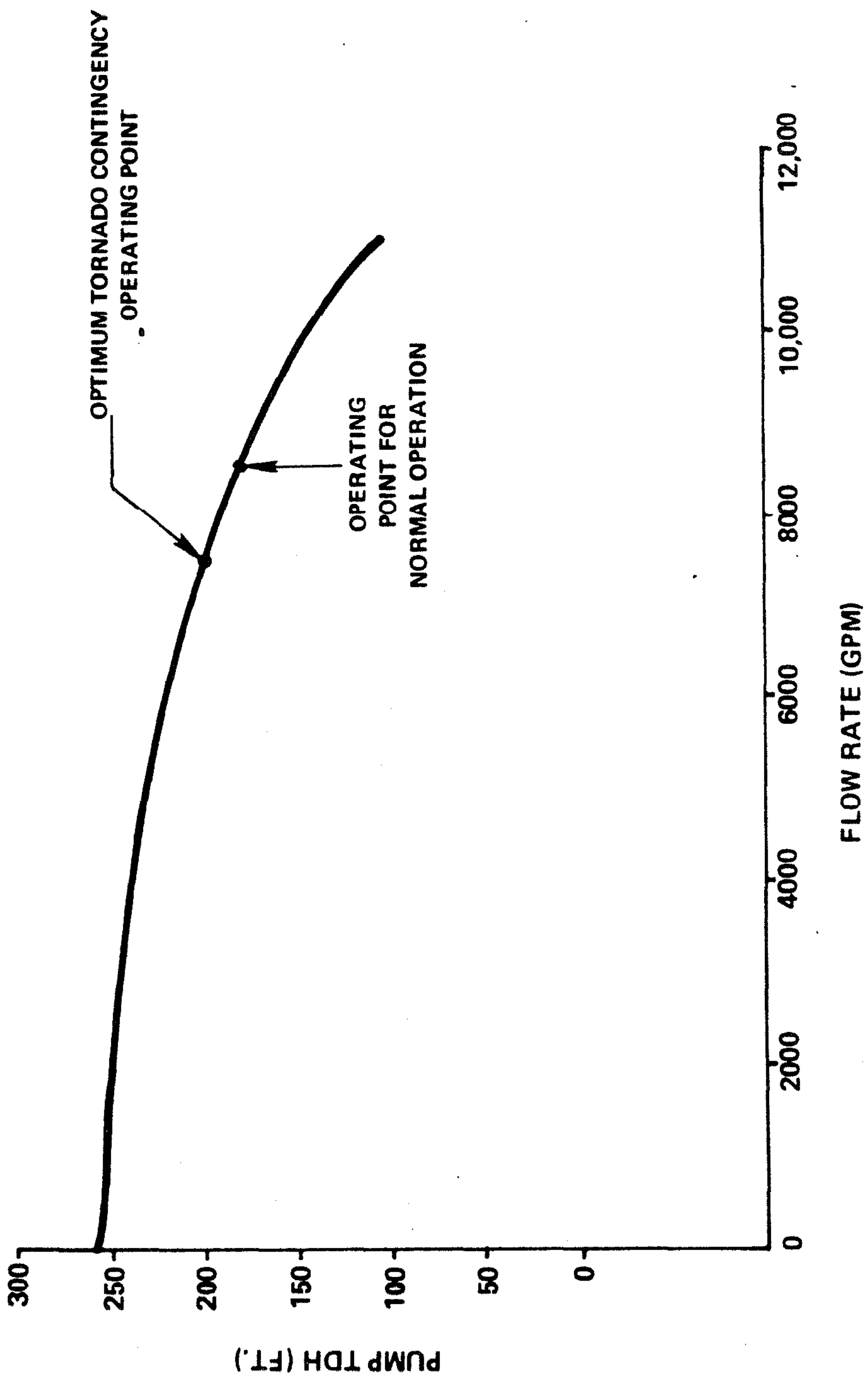
RADIAL DISPLACEMENT OF
SHELL ELEMENTS IN IMPACT ZONE

FIGURE 3F-16



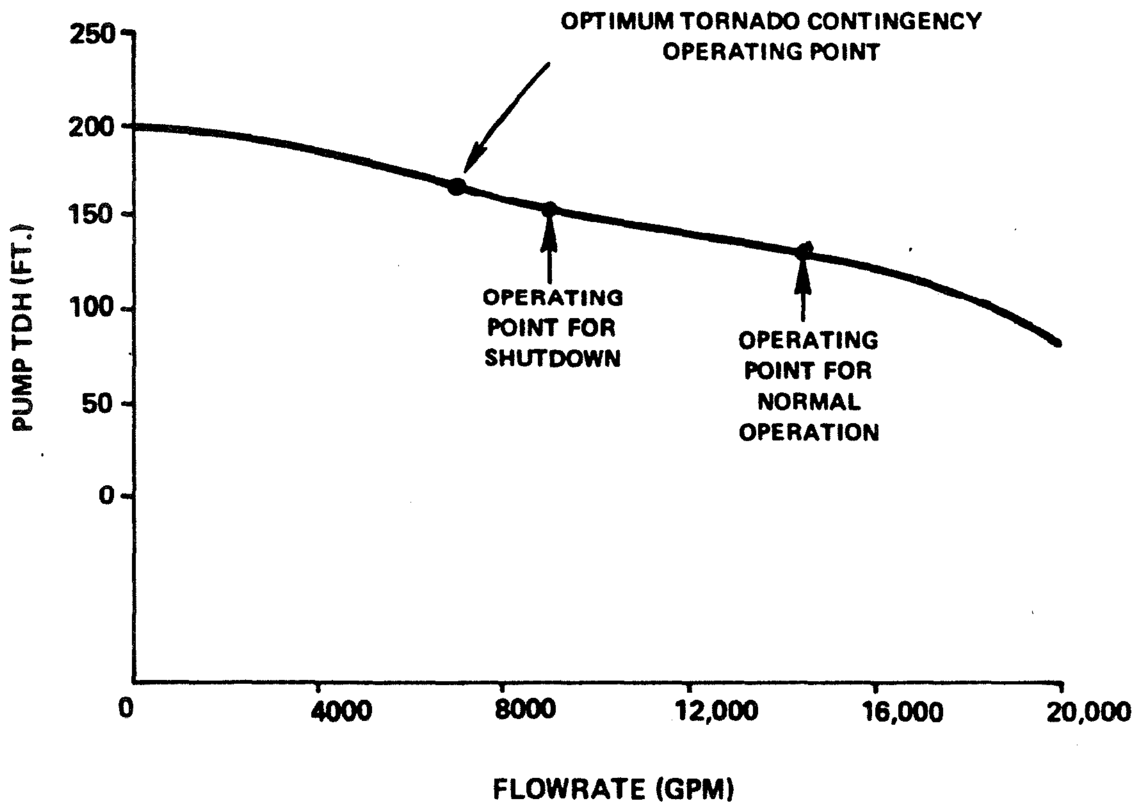
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FLOW AND HEAD CHARACTERISTICS
AT CCW HEAT EXCHANGER FOR
TORNADO CONTINGENCY OPERATING MODE
FIGURE 3F-17



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

COMPONENT COOLING WATER PUMPS
PERFORMANCE CURVES
FIGURE 3F-18



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

INTAKE COOLING WATER PUMPS
PERFORMANCE CURVES
FIGURE 3F-19

APPENDIX 3G - PIPING PENETRATIONS CALCULATIONS

APPENDIX 3G1

REPORT NO. 71983-1

Information provided in these appendices summarizes original analyses and is maintained for documentation and traceability purposes.

TUBE TURNS
LOUISVILLE 1, KENTUCKY

ENGINEERING DEPARTMENT REPORT NO. 71983-1

FLUED HEAD CALCULATIONS FOR HUTCHINSON ISLAND PENETRATIONS

INTRODUCTION

The purpose of this report is to summarize the stresses in the Type I and Type III flued heads under a combination of pipe rupture and thermal expansion loading as specified by Ebasco. The stresses to be presented are on a primary stress basis consistent with the requirements of ASME Section III for faulted conditions. The basic equations used are developed in Tube Turns Engineering Department report no. 5,114 A, attached. Some additional equations are needed for Type I heads. These are developed as needed in the text of this report.

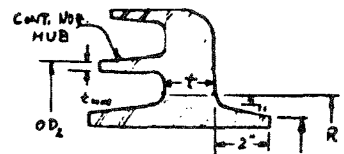
ASSUMPTIONS

1. Load combinations to be considered for Type III are:
   - A. Bending and Axial
   - B. Bending and Transverse (Shear)
   - C. Torsion and Transverse (Shear)
2. Allowable stresses are taken as material yield at process pipe operating temperature. The specific numerical values will be taken from the 1971 edition of Section III.
3. The stresses in the process pipe tapered hubs need not be evaluated. By virtue of a tapered hub, additional thickness is provided which obviously results in lower primary stresses than the attached piping.
4. For Type III heads, the loads are resisted by the containment nozzle hub.
5. For Type I heads, the loads are resisted as follows:
   - A. Axial and Torsion by Trunnions
   - B. Shear and Bending by two shear plates located 7 feet apart.
   - C. The trunnions are assumed loaded at a point 4 in. outward from the flued head.
6. For Type I heads, load combinations are as specified by Ebasco. Informal memo dated December 22, 1971.

TUBE TURNS
LOUISVILLE 1, KENTUCKY

71983-1

BASIC DIMENSIONAL DATA FOR TYPE III HEADS



$$R = \frac{OD_1}{2} + 2 \times \tfrac{1}{4} = \frac{OD_1}{2} + .5$$

$$t_N = t_{min} \text{ PER DWG}$$

$$D_m = OD_2 - t_N$$

OD_1 = PIPE O.D. OR SOCKET O.D. PER DWG

PEN	OD_1(in)	R(in)	OD_2(in)	t_N(in)	D_m(in)	t(in)	Material	Proc. Pipe Temp(°F)
5	3.062	2.031	10.750	.519	10.231	2	A105 GrII	520
6	3.062	2.031	10.750	.519	10.231	2	A105 GrII	520
26	3.312	2.156	12.750	1.148	11.602	3	A182 F304	450
27	3.312	2.156	10.750	1.125	9.625	2 1/2	A182 F304	120
28	.875	.938	6.625	.432	6.193	1 1/2	A182 F304	550
36	6.625	3.812	16.000	1.250	14.750	3 1/2	A182 F304	330
37	6.625	3.812	16.000	1.250	14.750	3 1/2	A182 F304	330
38	6.625	3.812	16.000	1.250	14.750	3 1/2	A182 F304	330
39	6.625	3.812	16.000	1.250	14.750	3 1/2	A182 F304	330
40	10.750	5.875	24.000	.765	23.235	3	A182 F304	300
41	3.062	2.031	8.625	.438	8.193	2	A182 F304	300
44	1.500	1.25	6.625	.628	5.997	1 1/2*	A182 F304	150
48	?	?	6.625	.628	5.997	1 1/2*	?	100
64	10.750	5.875	24.000	.765	23.235	3	A182 F304	300

* Estimated due to incomplete load versus dimension data

TUBE TURNS
LOUISVILLE 1, KENTUCKY

71983-1

LOADS DATA FOR TYPE III HEADS

PEN	AXIAL	TRANS.	BENDING	TORSION
5	4100lb	4100lb	76000 in. lb.	76000 in. lb.
6	4100	4100	76000	76000
26	5700	5700	107500	107500
27	8600	8600	107500	107500
28	600*	600*	1000*	1500*
36	14300	14300	700,000	1775000
37	14300	14300	700,000	1,775,000
38	14300	14300	700,000	1,775,000
39	14300	14300	700,000	1775000
40	34300	34300	2,336,000	2,336,000
41	1630	1630	51300	51300
44	?	?	?	?
48	9000*	9000*	10,000*	10,000*
64	34300	34300	233600	233600

*Use design basis earthquake plus thermal as faulted condition due to negligible or lack of rupture loads.

The load data tabulated above was computed by combining thermal loads and pipe rupture loads as supplied by Ebasco in the December 22, 1971 informal memo.

STRESS RESULTS FOR TYPE III HEADS

The equations used and method of determining "Maximum Stress Intensity" is provided in report 5.114 A attached. The calculated results are presented in tables 1 and 2 attached. It should be noted that all stresses are well within Code allowables thereby justifying the head design for pipe rupture loading.

TUBE TURNS
LOUISVILLE 1, KENTUCKY

71983-1

TYPE III HEAD BODY STRESSES

	STRESSES FOR INDIVIDUAL LOADS				MAXIMUM STRESS INTENSITIES			
PEN	Axial τ_{RZ} (PSI)	Bending τ_{RZ} (PSI)	Trans. σ_R (PSI)	Torsion $\tau_{R\theta}$ (PSI)	Ax + Ben. S(PSI)	Trans + Bend S(PSI)	Trans + Tor. S(PSI)	Allow S_y (PSI)
5	161	2932	322	1461	6186	5873	2940	28100
6	161	2932	322	1461	6186	5873	2940	28100
26	273	2454	446	1227	5354	4928	2494	20050
27	212	2945	424	1473	6314	5905	2976	28000
28	68	241	136	181	618	504	387	18800
36	170	4381	340	5555	9102	8769	11115	21400
37	170	4381	340	5555	9102	8769	11115	21400
38	170	4381	340	5555	9102	8769	11115	21400
39	170	4381	340	5555	9102	8769	11115	21400
40	310	7181	620	3591	14982	14375	7209	22500
41	64	1979	128	990	4086	3960	1984	22500
44	No Computations Due to Incomplete Load							27500
48	And Dimensional Data							
64	310	7181	620	3591	14982	14375	7209	22500

TABLE I

T-766

TYPE III CONTAINMENT NOZZLE HUB STRESSES

	STRESSES FOR INDIVIDUAL LOADS				MAXIMUM STRESS INTENSITIES			
PEN	Axial σ_R(PSI)	$\Theta = 0°$ Bending σ_a (PSI)	$\Theta = 90°$ Transverse $\tau_{r\theta}$(PSI)	Torsion $\tau_{r\theta}$(PSI)	Ax + Ben. S(PSI)	Trans + Ben S(PSI)	Trans + Tor. S(PSI)	Allow S_y(PSI)
5	246	1781	492	891	2027	1781	2766	28100
6	246	1781	492	891	2027	1781	2766	28100
26	136	886	272	443	1022	886	1430	20090
27	253	1313	506	657	1566	1313	2326	28000
28	71	77	142	58	148	284	400	18800
36	247	3277	494	4155	3524	3277	9298	21400
37	247	3277	494	4155	3524	3277	9298	21400
38	247	3277	494	4155	3524	3277	9298	21400
39	247	3277	494	4155	3524	3277	9298	21400
40	614	7202	1228	3601	7816	7202	9658	22500
41	145	2222	290	1111	2367	2222	2802	22500
44			NO COMPUTATIONS DUE TO LACK OF LOAD DATA					27500
48	761	564	1522	282	1325	3044	3608	
64	614	7202	1228	3601	7816	7202	9658	22500

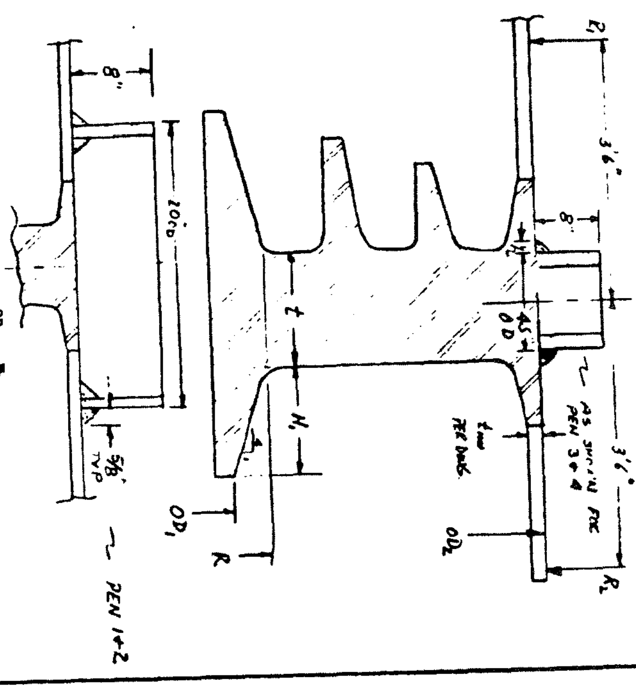
TABLE 2

71983-1

TUBE TURNS, LOUISVILLE 1, KENTUCKY

T-766

BASIC DIMENSIONAL DATA FOR TYPE I HEADS



$R = \frac{OD}{2} + \frac{H}{4}$

t_N = $t_{N MIN}$ Per drawing

$D_N = OD_2 - t_N$

Mat'l: SA105 GRII

PEN	OD(in)	H(in)	R(in)	OD_2(in)	P	t_N(in)	D_N(in)	t(in)	OP. TEMP(°F)
1+2	35.5	4	18.75	72		1.094	70.906	6.5	520
3+4	20	3	10.75	58		.875	57.125	5	440

71983-1

TUBE TURNS, LOUISVILLE 1, KENTUCKY

LOAD DATA FOR TYPE I PENETRATIONS 71983-1

	AXIAL	TRANS.	BENDING	TORSION	CASE
MAIN STEAM CASES	346,000 lb.	— lb	1,200,000 in lb	5,400,000 in lb	1
	—	191,000	49,200,000	5,400,000	2
	—	80,000	28,800,000	47,400,000	3
	236,000	—	28,800,000	5,400,000	4
FEEDWATER CASES	19,000	60,000	2,640,000	180,000	1
	—	86,000	5,040,000	120,000	2
	—	60,000	2,640,000	120,000	3
	—	86,000	2,640,000	2,520,000	4

Here again the above combined pipe rupture and thermal expansion load data was obtained from the Ebasco memo of December 22, 1971.

DEVELOPMENT OF ADDITIONAL EQUATIONS FOR TYPE I

The equations developed in report 5.114 A for head body stresses are still directly applicable to Type I heads. The nozzle or hub stress equations are also still valid. However, some development of the load input to the nozzle equations is required. This will be performed in the following. In addition, the equations for computing the loads and subsequently the weld attachment stresses for the trunnions will be developed below.

1. Nozzle Loads

Per assumptions stated above, the outermost hubs of the flued head resist the applied bending moment (M) and transverse load (V). Due to symmetry, each hub resists one half of the bending moment, and also one half the applied transverse load. Therefore, using the equations of statics, the hub loads become:

$$\left.\begin{aligned} M_H &= \frac{1}{2}M + \frac{1}{2}VL \\ V_H &= \frac{M}{2L} + \frac{1}{2}V \end{aligned}\right\} \quad \text{Where } L = 42 \text{ in.}$$

Using M_H and V_H above, the stress intensities in the nozzle hubs can be determined in a similar manner to the Type III heads.

2. Trunnion Loads

Per assumptions stated above, the two trunnions resist the axial load (F) and torsional load (Mt). Therefore, the loads transmitted to the trunnions become:

$$F_A = \frac{1}{2}F$$

$$F_t = \frac{Mt}{OD_2 + 8}$$

$$F_R = \sqrt{F_A^2 + F_t^2}$$

3G-5

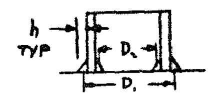
The loads transmitted to the trunnion attachment welds become:

$$V_o = F_R$$
$$M_o = 4F_R$$

Since the trunnion wall thickness is greater than the total weld throat, the weld stresses will govern. Therefore, only they will be evaluated below:

$$\tau_{AVG} = \frac{V_o}{.707\pi h(D_1 + D_2)}$$

$$\sigma_b = \frac{M_o}{I/C}$$



Where $I = \pi R^3 t$ for thin ring

D_1 = PIPE OD + .5h

D_2 = PIPE ID - .5h

Therefore

$$I = \pi \times .707h \left\{ \left(\frac{D_1}{2}\right)^3 + \left(\frac{D_2}{2}\right)^3 \right\}$$
$$C = D_1/2$$

The above bending and shear stress are combined to form a "Maximum Stress Intensity" (as outlined in report 5.114 A) and compared to the yield strength at operating temperature for the lower of head material or trunnion material.

STRESS RESULTS FOR TYPE I HEADS

The head body stress results are tabulated in Table 3 while the nozzle stresses are shown in Table 4. The trunnion weld stresses are shown below:

A. MAIN STEAM (P1 and P2)

By inspection case 3 is limiting

$F_A = 0$

$F_t = \frac{47,400,000}{72 + 8} = 592,500$

$F_R = 592,500$ lb

$V_o = 592,500$ lb

$M_o = 4 \times 592,500 = 2,370,000$ in. lb.

$$\tau_{AVG} = \frac{592,500}{.707\pi \times .625\,(20.31 + 16.625)} = 11251 \text{ PSI}$$

$$I = \pi \times .707 \times .625 \left\{ \left(\frac{20.31}{2}\right)^3 + \left(\frac{17.625}{2}\right)^3 \right\} = 2404 \text{ in.}^3$$

$$C = \frac{20.31}{2} = 10.16$$

TUBE TURNS
LOUISVILLE 1, KENTUCKY

$\sigma_b = \frac{2{,}370{,}000 \times 10.16}{2404} = 10016$ PSI

$S = 2\sqrt{\left(\frac{10016}{2}\right)^2 + (11251)^2} = \underline{24630 \text{ PSI}}$

$S_y = 27800$ PSI

B. FEEDWATER (F3 and F4)

By inspection Case 4 is limiting

$F_A = 0$

$F_t = \frac{2{,}520{,}000}{66} = 38182$

$F_R = 38182$ lb.

$V_0 = 38182$ lb.

$M_0 = 4 \times 38182 = 152730$ in. lb.

$\tau_{ave} = \frac{38182}{.707\pi \times .5(4.75)} = 7238$ PSI

$I = \pi \times .707 \times .5\left(\frac{4.75}{2}\right)^3 = 14.877$

$c = \frac{4.75}{2} = 2.375$

$\sigma_b = \frac{152730 \times 2.375}{14.877} = 24382$

$S = 2\sqrt{\left(\frac{24382}{2}\right)^2 + (7283)^2} = \underline{28157 \text{ PSI}}$

$S_y = 29300$ PSI

Examination of the stress intensities in Tables 3 and 4 as well as those above demonstrate an acceptable design for the pipe rupture conditions specified.

Continued on Page 10

TUBE TURNS
LOUISVILLE 1, KENTUCKY

TYPE I - HEAD BODY STRESSES

PEN	CASE	AXIAL τ_{an} (PSI)	BENDING τ_{bn} (PSI)	TRANS V_n (PSI)	TORSION τ_θ (PSI)	MAX STRESS INTENSITY S(PSI)	ALLOWABLE S_T (PSI)
1 and 2	1	452	167	—	376	1448	28600
1 and 2	2	—	6853	499	376	13736	28600
1 and 2	3	—	4012	209	3301	10393	28600
1 and 2	4	308	4012	—	376	8675	28600
3 and 4	1	56	1454	355	33	3042	30180
3 and 4	2	—	2776	509	33	5576	30180
3 and 4	3	—	1454	355	33	2930	30180
3 and 4	4	—	1454	509	694	3262	30180

TABLE 3

TUBE TURNS
LOUISVILLE 1, KENTUCKY

TYPE I - NOZZLE HUB STRESSES

PEM	CASE	M_H (in. lb)	V_H (lb)	BENDING σ_z(PSI)	TRANS τ_{rz}(PSI)	STRESS INTENSITY	S_m(PSI)
1 and 2	1	600,000	14,286	139	117	234	28,600
1 and 2	2	28,611,000	681,214	6,623	5,591	11,182	28,600
1 and 2	3	16,080,000	382,857	3,722	3,142	6,284	28,600
1 and 2	4	14,400,000	342,857	3,333	2,814	5,628	28,600
3 and 4	1	2,580,000	61,429	1,150	782	1,564	30,180
3 and 4	2	4,326,000	103,000	1,929	1,312	2,624	30,180
3 and 4	3	2,580,000	61,429	1,150	782	1,564	30,180
3 and 4	4	3,126,000	74,428	1,394	948	1,896	30,180

TABLE 4

TUBE TURNS
LOUISVILLE 1, KENTUCKY

DISCUSSION

It can be easily observed that the stresses calculated herein are for the most part quite low in comparison to allowables. This condition results from proportioning the heads to produce Code acceptable stress values for normal operating loads as well as pipe rupture loads. Under normal operating conditions the designer must consider all stresses in the head including the effects of structural discontinuity and stress concentration which are not required for faulted analysis.

DATE: 29 FEB 72 WRITTEN BY: W S Haberman
William S. Haberman

APPROVED BY: Michael V. Malkmus
Michael V. Malkmus

APPENDIX 3G2

REPORT NO. 71983-3

TUBE TURNS
LOUISVILLE 1, KENTUCKY

Rev. 1

REPORT NO. 71983-3

Design Calculations

for

Florida Power & Light Company

Hutchinson Island Plant

Type III Containment Piping Penetration Assemblies

P.O. No. NY-422264

Included are:

Penetration No. 5, Blowdown (SG-1A), I-2B-1
" " 6, Blowdown (SG-1B), I-2B-2
" " 26, Letdown Line, I-2CH-103
" " 27, Charging Line, I-2CH-109

" " 36, Safety Injection Loop 1A2, I-6SI-113
" " 37, " " " 1A1, I-6SI-112
" " 38, " " " 1B1, I-6SI-111
" " 39, " " " 1B2, I-6SI-110
" " 40, Shutdown Cooling, I-10SI-422
" " 41, Safety Injection Tank Test, I-2SI-479
" " 44, Reactor Coolant Pumps Bleed Off, I-3/4 CH-12?
" " 64, Shutdown Cooling, I-10SI-420

Prepared By: W. S. Haberman
W. S. Haberman
Bellows Engineering
Date: May 8, 1972
1 Oct 23, 1972

Approved By: M. V. Malkmus
M. V. Malkmus
Mgr. Bellows Engineering
Date: May 8, 1972

3G-9

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO. 71983-3

TABLE OF CONTENTS

1. Summary of Report

As substantiated by the calculations herein, all components in this group of penetration assemblies have been demonstrated to comply with the design information supplied. The only exception are the flued heads which are evaluated as a group using four typical heads. This is treated in Report No. 71983-5.

In an effort to evaluate the penetration assemblies as a whole, we have considered the following major areas of concern:

1. Process Pipe pressure carrying integrity.
2. Pipe rupture integrity
3. Bellows integrity
4. Jet Impingement integrity
5. Seismic capability.

In evaluating the above we have utilized the allowable stresses from USAS B31.7-1969 for Class II piping for all but faulted conditions where we have used minimum pipe yield at process pipe operating temperature from ASME Section III (1971) due to a lack of adequate guidance from B31.7

2. Bellows Design Calculations

A. General: The intent of this section is to demonstrate by suitable computation, the structural adequacy of the secondary seal bellows for the design conditions stated in Paragraph B below. The configuration of the bellows is depicted on Drawings 71983 D3.1 and 71983 C3.2 attached.

B. Design Conditions:

Pressure = ± 5 PSIG

Temperature = 264°F

Material = A240 TP 316L

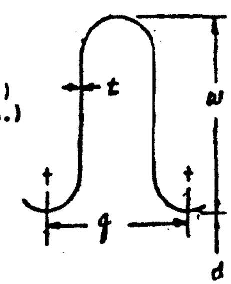
Movements:	#1 600 Cycles	#2 7000 Cycles
Axial Extension	2.5 in.	.5
Lateral	± .5	± .125

C. References:
1. USAS B31.7-1969
2. Standards of the Expansion Joint Manufacturers' Assn., Inc. Third Edition 1969.

D. Bellows Hoop Stress: The bellows hoop or circumferential membrane stress is determined by use of the Barlow Formula for straight pipe suitably modified to account for a convoluted distribution of metal. As applied to a bellows the equation takes the following form:

$$S_b = \frac{Pd}{2t\,n_p}\;\frac{\left(\frac{d+w}{d}\right)^{3/2}}{(.571 + 2w/q)}$$

WHERE:
S_b = Stress (PSI)
P = Design Pressure (PSIG)
d = Bellows Root Diameter (in.)
t = Bellows Wall Thk-1 Ply (in.)
w = Convolution Height (in.)
q = Convolution Pitch (in.)
n_p = No. Bellows Plies



E. Meridional Bending Pressure Stress

The bellows meridional stress (stress tending to crush or collapse the convolutions in an axial direction) is determined by the following theoretical formula, with empirical modifications to correlate with test data.

$$S_m = \frac{Pw^2}{10\ 2/3\ t^2\ n_p}$$

F. Bellows Movement Capacity: The movement capability of a bellows is evaluated on an "equivalent axial movement" per convolution basis. For the secondary seal bellows, the allowable movements have been limited to the following:

Allowable compressive movement ($e_{comp.}$)

$$= .85(q/2 - t\ n_p)$$

Allowable extention movement ($e_{ext.}$)

$$= .43(q/2 - t\ n_p)$$

The allowable for compression is based primarily on physical limitation from the geometry. For extension, the limitation is based on empirical data to prevent dimpling of the convolution crest and to prevent excessively high stresses that would significantly reduce cyclic life.

The axial and lateral movements tabulated in Paragraph B are converted to equivalent axial movement per convolution by the following formulas: (Ref.: Standards of the Expansion Joint Manufacturers' Association):

$e_x = \frac{x}{n} =$ Axial Traverse per Conv. (+ is compressive)

$e_y = \frac{3Dy}{n(L-X)} =$ Equivalent Axial Traverse per Conv. (caused by Lateral Movement)

$e_{pc} = \frac{X_{pc}}{n} =$ Axial Traverse due to Precompression (+ is compressive)

$e_{ext} = -e_x - e_{pc} + e_y =$ Total Axial Ext. per Convolution

WHERE: X = Axial Movement (in.)
y = Lateral Movement (in.)
X_{pc} = Precompression (in.)
D = Conv. Crest Dia. = d + 2w (in.)
n = No. Convolutions
L = Convoluted Length = $nq - X_{pc}$ (in.)

G. Bellows Spring Rates: The bellows axial spring rate has been proven experimentally to be predicted by the following equation:

$$f_x = 2\, n_p(d+w)\ E\ \gamma^3/Cu$$

WHERE: f_x = Bellows axial spring force per convolution (Lb/In.)
E = Modulus of Elasticity $\times 10^{-6}$ (PSI) = 27.4
$\gamma = \frac{100t}{w}$ = Thickness to height ratio.
Cu = Spring rate denominator - an empirical factor which is a function of convolution pitch, height and diameter.
n_p = Number of bellows plies.

The axial spring rate of the total joint can be expressed as:

$$kx = \frac{Fx}{x}$$

WHERE: $F_x = fx\ e_x$

$e_x = \frac{x}{n}$

x = 1 in.

Therefore by substitution we obtain:

$$kx = \frac{fx}{n} \quad \text{(Lb/In.)}$$

The lateral spring rate can be expressed as:

$$k_y = \frac{Fy}{y}$$

WHERE $F_y = \frac{fx\ D\ ey}{2L}$

$e_y = \frac{3\ Dy}{n\ (L - X)}$

y = 1 in.

Therefore by substitution we obtain:

$$k_y = \frac{f_x D}{2L} \cdot \frac{3D}{n(L-\chi)}$$

$$k_y = \frac{3 f_x D^2}{2Ln(L-\chi)} \quad (\text{Lb./In.})$$

H. Cyclic Life Determination: It has been proven through extensive testing that the cyclic life of a U-shaped bellows can be predicted by use of the following formula:

$$N = 4\left(\frac{\theta S_u}{S_o'}\right)^4$$

WHERE: N = CALCULATED NUMBER OF CYCLES

S_u = ULTIMATE TENSILE STRENGTH OF BELLOWS MATERIAL (PSI)

$S_o = (1 + P/n_p p')^2\, 675\, \tau E e_o / w$ = TRAVERSE STRESS PARAMETER (PSI)

$S_p = 4.5\, C_u^4 P / (1 + P/n_p p')\, \tau^2 n_p$ = PRESSURE STRESS PARAMETER (PSI)

$S_o' = S_o + S_p$ = TOTAL STRESS PARAMETER (PSI)

$p' = S_c t / 2w_e$

$\tau = \frac{100t}{w}$ = THICKNESS TO HEIGHT RATIO

$E = 27.4$ = MODULUS OF ELASTICITY $\times 10^{-6}$ (PSI)

$e_o = e_{cyc} - e_e/3$ = EFFECTIVE TRAVERSE PER CONVOLUTION (IN.)

w = CONVOLUTION HEIGHT (IN.)

S_c = 17500 PSI = COLD ALLOWABLE BELLOWS STRESS

S_H = 16400 PSI = HOT ALLOWABLE BELLOWS STRESS (264°F)

$w_e = w - t/3$ = EFFECTIVE CONV. HT (IN)

n_p = NUMBER OF BELLOWS PLIES

$\theta = \frac{S_c + S_H}{2S_c}$ = HOT/COLD STRESS RANGE RATIO $= .97$

$e_e = \frac{w_e^2}{450\tau}$ = ELASTIC TRAVERSE PER CONVOLUTION (IN.)

For purposes of cyclic life determination, the non-cyclic precompressive movement installed at the factory is not considered as part of the total equivalent axial movement per convolution. Therefore, we obtain:

$$e_{cyc} = -e_x + e_y$$

I. RESULTS:

1. The static capacity evaluation of the various secondary seal bellows from both a pressure stress and movement standpoint is tabulated in Table 1, page 10. The allowable stress has been taken from B31.7 Appendix A, Table A.8 while the allowable bellows equivalent axial traverse is computed per Paragraph F of this report. For completeness pertinent bellows dimensions have also been included in the table.

2. Computation of bellows axial and lateral spring rates are included in Table 2 page 11.

3. Computation of bellows cyclic life is included in Table 3 page 12. Since the 600 cycle design movements (#1) exceed 7000 calculated cycles it was unnecessary to calculate the life of the lesser condition (#2).

J. Discussion: As can be easily observed in the results presented in Tables 1 and 3, all bellows have more than adequate capacity to meet design objectives. Finally, in summary it is worth mentioning a few words relative to the formulas used in bellows design at Tube Turns. Due to the vast complexity involved in a rigorous exacting analytic approach to the problem which still must be correlated with test data to be useful, we have adopted the procedure of using empirical modifications to a somewhat more basic theoretical approach. These empirical correlations have been obtained through hundreds of tests on actual production bellows. The net result is a procedure with good experimental correlation at a large expenditure of effort. For this reason we have treated this information in a proprietary manner. However, it is available for review in our offices.

BELLOWS STATIC CAPACITY RESULTS

PEN NO.	d (IN)	t (IN)	w (IN)	q (IN)	n	X_{pc} (IN)	S_b (PSI)	S_m (PSI)	S_{allow} (PSI)	$x=2.3$ e_x (IN)	$y=.75$ e_y (IN)	e_{pc} (IN)	e_{ext} (IN)	e_{allow} (IN)
5	13.19	.031	1.375	1.0	15	2.375 →	187	460	17500 →	-.153	.160	.158	.155	.188
6	13.19	.031	1.375	1.0	15		187	460		-.153	.160	.158	.155	.188
26	15.19	.031	1.375	1.0	16		214	460		-.144	.159	.149	.154	.188
27	13.19	.031	1.375	1.0	15		187	460		-.153	.160	.158	.155	.188
36-39	19.25	.050	2.0	1.5	12		180	375		-.192	.244	.198	.238	.279
40,64	27.25	.050	2.0	1.5	13		243	375		-.177	.278	.183	.272	.279
41	11.94	.031	1.375	1.0	15		173	460		-.153	.148	.158	.143	.188
44	9.94	.031	1.375	1.0	14		150	460		-.164	.147	.170	.141	.188

TABLE 1

AXIAL AND LATERAL BELLOWS SPRING RATES

PEN NO.	d (IN.)	t (IN)	ω (IN.)	g (IN.)	n	X_{pc} (IN.)	τ	C_u	f_x (LB/IN/KNOT)	L (IN)	1-x (IN)	k_x (LB/IN)	k_y (LB/IN.)
5	13.19	.031	1.375	1.0	15	2.375	2.25	2.55	7100	12.63	14.93	470	960
6	13.19	.031	1.375	1.0	15		2.25	2.55	7100	12.63	14.93	470	960
26	15.19	.031	1.575	1.0	16		2.25	2.60	7900	13.63	15.93	490	1100
27	13.19	.031	1.375	1.0	15		2.25	2.55	7100	12.63	14.93	470	960
36-39	19.25	.050	2.0	1.5	12		2.50	2.59	14200	15.63	17.93	1200	3400
40,64	27.25	.050	2.0	1.5	13		2.50	2.69	18600	17.13	19.43	1400	6300
41	11.94	.031	1.375	1.0	15		2.25	2.47	6000	12.63	14.93	400	680
44	9.94	.031	1.375	1.0	14		2.25	2.41	5800	11.63	13.93	410	620

TABLE 2

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO. 71983-3

SEAL BELLOWS CYCLIC LIFE RESULTS

PEN NO.	d (IN.)	t (IN.)	ω (IN.)	q (IN.)	n	X_{pc} (IN)	C_u	$X=-2.3$ e_x (IN.)	$y=.76$ e_y (IN.)	e_{cyc} (IN.)	S_o (PSI)	S_p (PSI)	S_o' (PSI)	N CALC	N MOD
5	13.19	.031	1.375	1.0	15	2.375	2.55	-.153	.160	.313	8750	90	8840	77000	600
6	13.19	.031	1.375	1.0	15		2.55	-.153	.160	.313	8750	90	8840		
26	15.19	.031	1.375	1.0	16		2.60	-.149	.159	.303	8440	100	8540		
27	13.19	.031	1.375	1.0	15		2.55	-.153	.160	.313	8750	90	8840		
36-39	19.25	.050	2.0	1.5	12		2.59	-.192	.244	.436	9350	80	9430		
40,64	27.25	.050	2.0	1.5	13		2.69	-.177	.278	.465	9800	100	9900		
41	11.94	.031	1.375	1.0	15		2.47	-.153	.148	.301	8380	80	8460		
44	9.94	.031	1.375	1.0	14		2.41	-.164	.147	.311	8690	80	8770		

TABLE 3

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO. 71983-3

3. Process Pipe Wall Thickness Check

A. General: The process pipe minimum wall requirements for design pressure and temperature as specified on the "Ebasco" drawings has been computed per the following equation from B31.7:

$$t_m = \frac{PD}{2\,(SE + .4P)}$$

The SE values in the above equation were taken from B31.7 Appendix A, Table A8. The required nominal thickness was taken as:

$$t_{NOM} = t_m/.875$$

B. References:
1. USAS B31.7 - 1969
2. Ebasco Dwg. No. 8770-6-213

C. Results:
The results of this evaluation are presented in Table 4 below. Note that all process pipe possess sufficient wall thickness to resist internal pressure.

TUBE TURNS
DIVISION OF CHEMETRON CORPORATION
LOUISVILLE 1, KENTUCKY

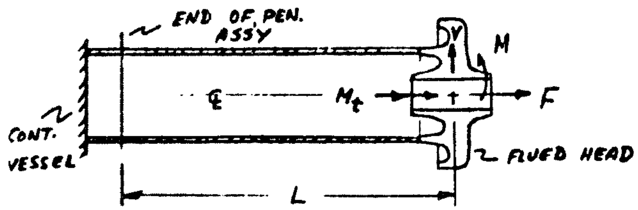
PROCESS PIPE REQUIRED WALL THICKNESS

PEN NO.	P (PSIG)	T (°F)	MAT'L	SE (PSI)	D_o (IN.)	CALC t_M (IN.)	REQ'D t_{min} (IN.)	ACT. t_{min} (IN.)
5	985	550	A106GRB	15000	2.375	.076	.087	.218
6	985	550	A106GRB	15000	2.375	.076	.087	.218
26	2485	550	A312TP304	14450	2.375	.192	.219	.344
27	2735	250	A312TP304	16050	2.375	.190	.217	.344
36-39	2485	650	A312TP304	14300	6.625	.540	.617	.719
40,64	300	350	A312TP304	15250	10.75	.105	.120	.250
41	300	350	A312TP304	15250	2.375	.0233	.027	.154
44	2485	250	A312TP304	16050	1.05	.0767	.088	.219

TABLE 4

4. Nozzle (Guard Pipe) Stresses Due to Pipe Rupture:

A. General: For purposes of analysis, consider the following "free body diagram" of a typical penetration assembly. It will be assumed that the bellows has little if any effect to this particular problem.



In the above M, F, V & M_t each represent a combined effect of thermal expansion and pipe rupture loading. This is arrived at by assuming one process hub to be undergoing its normal piping flexibility thermal expansion reactions while the other hub is loaded by pipe rupture reactions. For conservatism these are assumed additive. For analysis purposes, the following load combinations will be considered:

1. Bending and Lateral (M, V)
2. Torsion and Lateral (M_t, V)
3. Bending and Axial (M, F)

By inspection of the loads and dimensions it becomes obvious that the maximum stress intensity occurs at the end of the penetration assembly and is due to either Case 1 or 2 above. To determine the stresses the following formulas will be used:

CASE 1

$$\sigma_x = \frac{M + VL}{Z}$$

$$S = \sigma_x$$

CASE 2

$$\tau = \frac{M_t}{2Z}$$

$$\sigma_x = VL/Z$$

$$S = 2\sqrt{\left(\frac{\sigma_x}{2}\right)^2 + \tau^2}$$

$$= \frac{1}{Z}\sqrt{(VL)^2 + M_t^2}$$

From inspection of the above Case 2 can only govern where $M_t > M$.

B. References:

1. USAS B31.7 - 1969
2. Ebasco informal memo dated December 22, 1971 L.J. Sas to W. S. Haberman.
3. "Formulas for Stress and Strain" by R. J. Roark.

C. Results: The results of the above computations are presented in Table 5 attached (p.17). Note that all stresses are well within the allowables also tabulated. These allowables are based on minimum yield at process pipe operating temperature as specified on Ebasco Drawing No. 8770-G-213. The use of pipe yield as an allowable is based on an assumed faulted condition and consistent with Section III requirements. This is invoked due to lack of definitive allowables for B31.7 Class II piping under faulted conditions. It should be noted, however, that a 1.5S allowable for Class II piping would be met in this case anyway.

NOZZLE STRESSES DUE TO PIPE RUPTURE

PEN NO	d (IN)	t (IN)	Z (IN³)	L (IN)	M (IN LB)	V (LB)	M_t (IN LB)	VL (IN LB)	M+VL (IN LB)	$\sqrt{(VL)^2+M_t^2}$ (IN LB)	S (PSI)	S_{allow} $=S_y$ (PSI)
5	10.750	.593	45.5	106.2	76000	4100	76000	435000	511,000	NOT REQ'D	11200	27100
6	10.750	.593	45.5	106.2	76000	4100	76000	435,000	511,000	"	11200	27100
26	12.750	1.125	109.9	107.7	107500	5700	107500	610,000	718,000	"	6530	27100
27	10.750	1.125	74.3	106.4	107500	8600	107500	915,000	1,023,000	"	13,800	31400
36-39	16.000	1.218	194.5	110.4	700000	14300	1775000	1,579,000	2,279,000	2,376,000	12200	25400
40,64	24.000	1.000	400	103.7	2336000	34300	2336000	3,560,000	5,896,000	NOT REQ'D	14700	33100
41	8.625	.500	24.52	106.2	51300	1630	51300	173,000	224,000	"	9140	30500
44	6.625	.718	17.81	104.8	—	—	—					31400

TABLE 5

5. Lateral Jet Force Calculation

A. General: The jet forces for a longitudinal break are determined per the "Ebasco" spec. as follows:

F_j = KPA

WHERE: F_j = Jet Force (Lb.)
K = Phase Factor (pg. 9 of spec.)
P = Operating pressure (psig)
A = Pipe Flow Area (in.^{2})

Per the "Ebasco" Specification.

Break Length (ℓ) = 2 x Pipe Dia. (D)

Break Area = A_f

Therefore, Break Width (c) = $A_f/2D = A_f/\ell$

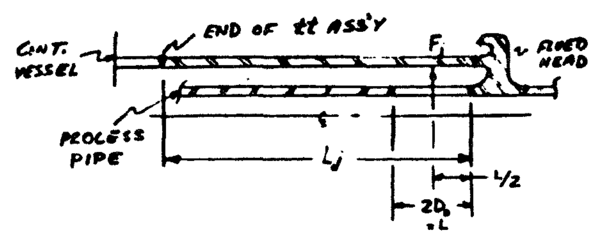
B. References:
1. Ebasco Specification No. FLO-8770.124
2. Ebasco Dwg. No. 8770-G-213

C. Results: The results of this computation using operating pressures and piping dimensions from Ebasco Dwg. No. 8770-G-213 are presented in Table 6 below. The forces will be used in the subsequent section.

6. <u>Nozzle Stresses due to Jet Impingement</u>

A. <u>General</u>: To determine the effect of a slot break within the penetration assembly, a conservative approach will be taken as follows:

1. The maximum bending stress that could occur in the nozzle would be at the left end of the assembly with a split at the right end of the process pipe as shown below:



$$M_{max} = F_j\,(L_j - L/2)$$

$$\sigma_{bx} = M_{max}/Z$$

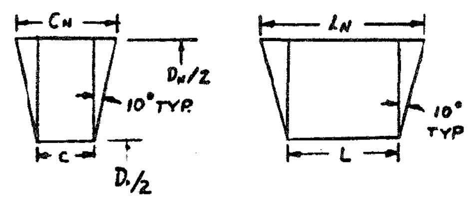
Note that stiffening afforded by process pipe to both the right and left of the head has been neglected. The values of F_j and L are taken from Table 6.

JET FORCES

PEN NO.	D_o (IN.)	t (IN.)	A_i (IN²)	P (PSIG)	FLOW MEDIA	K	L (IN.)	C (IN.)	F_j (LB.)
5	2.375	.218	2.953	885	STEAM	.63	4.75	.622	1640
6	2.375	.218	2.953	885	STEAM	.63	4.75	.622	1640
26	2.375	.344	2.240	2200	WATER	.97	4.75	.472	4970
27	2.375	.344	2.240	2335	WATER	.97	4.75	.472	5080
36-39	6.625	.719	21.1	2235	WATER	.97	13.25	1.59	45700
40,64	10.750	.250	82.5	300	WATER	.97	21.5	3.84	24000
41	2.375	.154	3.356	250	WATER	.97	4.75	.729	815
44	1.05	.219	.296	150	WATER	.97	2.10	.141	43.1

TABLE 6

2. The second area of concern is the hoop stress caused in the nozzle or guard pipe due to the jet force striking a finite area. The area is developed as follows consistent with the "Ebasco" spec:



$$C_N = C + 2\,\mathrm{TAN}\,10^\circ \left(\frac{D_N}{2} - \frac{D_o}{2}\right)$$

$$= C + (D_N - D_o)\,\mathrm{TAN}\,10^\circ$$

$$L_N = L + (D_N - D_o)\,\mathrm{TAN}\,10^\circ$$

$$A = C_N L_N$$

$$= [C + (D_N - D_o)\,\mathrm{TAN}\,10^\circ][L + (D_N - D_o)\,\mathrm{TAN}\,10^\circ]$$

$$P = \frac{F_j}{A}$$

$$\sigma_t = \frac{P(D_N + t)}{2t}$$

WHERE: D_N = Nozzle I.D. (in.)
D_o = Process Pipe (O.D.)
t = Nozzle Thickness (in.)
F_j = Jet Force (Lb.)
c = Slot Width (in.)
L = Slot Length (in.)

TUBE TURNS
DIVISION OF CHEMETRON CORPORATION
LOUISVILLE 1, KENTUCKY

REPORT NO. 71983-3

B. <u>References</u>:
1. Ebasco Spec. No. FLO-8770.124
2. "Formulas for Stress and Strain" by R. J. Roark.
3. USAS B31.7 - 1969

C. <u>Results</u>: The tabulated results of the above computations are presented in Tables 7 and 8 attached (pages 23 and 24). This condition has been assumed as faulted thereby justifying a specified minimum yield allowable stress. See Paragraph 4C for discussion on use of this allowable.

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO. 71983-3

NOZZLE BENDING STRESSES
DUE TO JET FORCES

PEN NO	F_J (LB)	L (IN)	L_J (IN)	Z (IN^3)	σ_B (PSI)	S_y (PSI)
5	1640	4.75	101.2	45.5	3560	27100
6	1640	4.75	101.2	45.5	3560	27100
26	4970	4.75	101.9	109.9	4500	27100
27	5080	4.75	101.2	74.3	6760	31400
36-39	45,700	13.25	104.2	194.5	22800	25400
40,64	24000	21.5	98.2	400	5250	33100
41	815	4.75	101.2	24.57	3280	30500
44	43.1	2.10	100.1	17.84	240	31400

TABLE 7

NOZZLE HOOP STRESSES DUE TO JET FORCES

PEN NO.	D_N (IN)	t (IN.)	D_o (IN.)	c (IN.)	L (IN.)	F_J (LB)	A (IN^2)	σ_t (PSI)	S_y (PSI)
5	9.564	.593	2.375	.622	4.75	1640	10.1	1390	27100
6	9.564	.593	2.375	.622	4.75	1640	10.1	1390	27100
26	10.500	1.125	2.375	.472	4.75	4970	11.8	2180	27100
27	8.500	1.125	2.375	.472	4.75	5080	9.04	2410	31400
36-39	13.564	1.218	6.625	1.59	13.25	45700	40.7	6810	25400
40,64	22.00	1.0	10.750	3.84	21.5	24000	137.	2020	33100
41	7.625	.500	2.375	.729	4.75	815	9.40	705	30500
44	5.189	.718	1.05	.141	2.10	43.1	2.46	72	31400

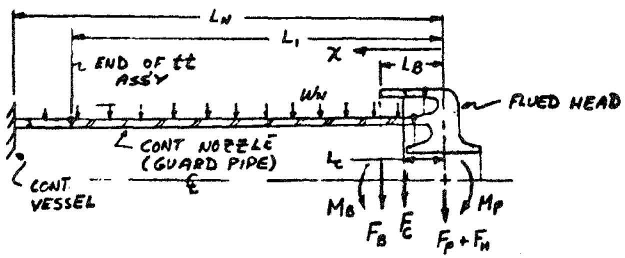
TABLE 8

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO. 71983-3

7. Penetration Seismic Analysis

A. General: Consider the following free body diagram of a typical Type III penetration less the bellows:



The general bending moment equation can be written as follows:

$$M_{Bx} = F_T x + M_T + \frac{w_N x^2}{2}$$

WHERE: $F_T = F_P + F_B + F_C + F_H$ — ①

$M_T = M_P - M_B - F_C L_C - F_B L_B$ — ②

F_P = D.B.E PIPING LOAD + THERMAL EXP. LOAD (LB.)

M_P = " " " " " " " (IN. LB.)

$F_C = \frac{W_C}{g} a$ = WELD END SEISMIC LOAD (LB.)

$F_H = \frac{W_H}{g} a$ = FLUED HEAD SEISMIC LOAD (LB.)

F_B = BELLOWS END REACTION (LB.)

M_B = " " " (IN. LB.)

$w_N = \frac{W_N'}{g} a$ = CONT. NOZZLE SEISMIC LOAD (LB/IN)

WJ
5 Apr 72

W_C = WELD END WGT. (LB)

W_H = FL. HD. WGT (LB)

W_N' = NOZZLE PIPE WGT/UNIT LENGTH (LB/IN.)

a = ACCELERATION (FT/SEC^2)

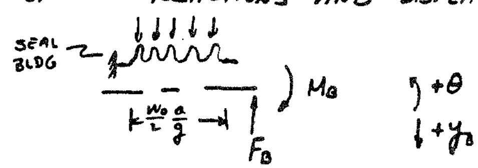
g = GRAVITATIONAL CONSTANT (FT/SEC^2)

THE MAXIMUM BENDING MOMENT WOULD OCCUR AT
$x = x_1$

THEREFORE, WE HAVE

$$M_{B_{L_1}} = F_T L_1 + M_T + \frac{W_N'}{2g} a L_1^2 \quad — ③$$

THIS WOULD BE A STRAIGHT FORWARD SOLUTION EXCEPT THE BELLOWS END REACTIONS ARE A FUNCTION OF THE DISPLACEMENTS IMPOSED. CONSIDER THE FOLLOWING BELLOWS FREE BODY DIAGRAM, AND RESULTING EQUATIONS OF REACTIONS AND DISPLACEMENTS:



$$F_B = \frac{W_B}{2}\frac{a}{g} - \frac{3 f_x D^2 g}{2 L^3}\left(y_B + \frac{\theta L}{2}\right) \quad — ④$$

$$M_B = \frac{W_B L}{12}\frac{a}{g} - \frac{3 f_x D^2 g}{4 L^2}\left(y_B + \frac{2\theta L}{3}\right) \quad — ⑤$$

NSH
5AK7.

WHERE: W_B = BELLOWS WGT (LB)

f_x = BELLOWS AXIAL SPRING RATE (LB/IN/CONV.)

D = BELLOWS CREST DIAMETER (IN)

δ = BELLOW CONV. PITCH (IN.)

L = BELLOWS CONVOLUTED LENGTH (IN.)

y_o = BELLOWS END DEFL. (IN.)

θ = BELLOWS END ROTATION (RAD.)

AT THIS POINT WE HAVE SUBSTITUTED ONE PAIR OF UNKNOWNS (F_B + M_B) FOR A SECOND SET (y_B + θ). THEREFORE, RELATIONS BETWEEN y_B + θ AND F_T, M_T, W_N ARE REQUIRED. THESE ARE:

$$\theta = -\frac{F_T L_N^2}{2EI} - \frac{M_T L_N}{EI} - \frac{W_N' a L_N^3}{12 g EI} \quad — ⑥$$

$$y_N = \frac{F_T' L_N^3}{3EI} + \frac{M_T L_N^2}{2EI} + \frac{W_N' a L_N^4}{16 g EI} \quad — ⑦$$

$$y_B = y_N + L_1 \theta \quad — ⑧$$

SUFFICIENT EQUATIONS HAVE NOW BEEN DEVELOPED TO ACHIEVE A SOLUTION. SEVERAL METHODS ARE AVAILABLE; HOWEVER, THE ONE TO BE USED HERE WILL BE AN ITERATIVE APPROACH. AS FOLLOWS

1. ASSUME $F_B = \frac{W_B}{2}\frac{a}{g}$ $M_B = \frac{W_B L}{12}\frac{a}{g}$
2. COMPUTE F_T AND M_T (EQ. ① + ②)
3. USING 2 COMPUTE y_B AND θ (EQ. ⑥, ⑦, ⑧)
4. USING 3 COMPUTE F_B AND M_B (EQ ④, ⑤)

3G-22

NSH
5AK7.

5. COMPARE 4 TO 1 IF A SIGNIFICANT DIFFERENCE IS PRESENT, ASSUME NEW VALUES OF F_B AND M_B
6. REPEAT 2 THRU 5 UNTIL CONVERGENCE IS ACHIEVED

AT THIS POINT SUFFICIENT INFORMATION IS NOW PRESENT TO COMPUTE THE MAXIMUM BENDING MOMENT IN THE NOZZLE PIPE (EQ. 3) AND HENCE THE MAXIMUM STRESS.

$$\sigma_b = \frac{M_{B_1}}{Z}$$

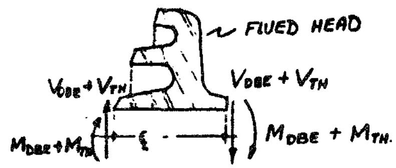
FUTHERMORE, END DEFLECTION INFORMATION RELATIVE TO THE BELLOWS IS AVAILABLE TO COMPUTE THE EQUIVALENT AXIAL TRAVERSE PER CONVOLUTION BY THE FOLLOWING FORMULA:

$$e = \frac{D\delta}{L}\left(\theta + \frac{3y_B}{L}\right) + \frac{W_B L a}{3 f_x D g}$$

B. REFERENCES:

1. "FORMULAS FOR STRESS AND STRAIN" BY R.J. ROARK
2. EBASCO INFORMAL MEMO DATED DECEMBER 22, 1971 L.J. SAS TO W.S. HABERMAN
3. EBASCO SPEC NO. FLO-8770.124
4. USAS B31.7 - 1969

C. Sample Calculation: For the purpose of illustrating the seismic capability of a complete penetration, P36 will be used as an example. First consider the head with applied piping reactions:



On the basis of the above, there would be no shear transmitted to the containment nozzle through the head. Therefore, for conservatism the shear on the left side will be neglected. Hence we have:

$$F_P = V_{DBE} + V_{TH} = 9000 + 9000 = 18000 \text{ LB.}$$

$$M_P = 2(M_{DBE} + M_{TH}) = 2(200000 \times 2) = 800000 \text{ IN LB}$$

To solve the problem the following additional input is needed.

$W_H = 500$ LB

$\left.\begin{array}{l} d_H = 16'' \\ t_H = 1.218'' \end{array}\right\} \therefore W_H' = \frac{192.3}{12} = 16$ LB/IN

$I = 1556 \text{ IN}^4$

$Z = I/c = 194.5 \text{ IN}^3$

$\left.\begin{array}{l} d_c = 20 \text{ IN.} \\ t_c = .500 \end{array}\right\} \therefore W_c = \frac{5}{12} \times 104.1 = 43.3$ LB

$L_B = 8.75$ IN. $\qquad L_1 = 110.4$

$L_c = 6.25$ IN. $\qquad L_H = L_1 + 9 = 119.4$ IN.

$\left.\begin{array}{l} D = 23.25 \text{ IN.} \\ g = 1.5 \text{ IN.} \\ f_x = 2 \times 7100 = 14200 \text{ LB/IN./CONV} \\ W_B = 110 \text{ LB} \\ L = 15.63 \text{ IN} \end{array}\right\}$ BELLOWS DATA

$$a = \sqrt{(a_{HOR})^2 + (a_{VERT.})^2}$$

$$= \sqrt{(2g)^2 + (.3g)^2}$$

$$= 3.61g \text{ FT/SEC}^2$$

$$F_c = 3.61 W_c = 3.61 \times 43.3 = 156 \text{ LB}$$

$$F_H = 3.61 W_H = 3.61 \times 500 = 1810 \text{ LB}$$

At this point we are ready to begin the iterative part of the solution.

1ST TRY

1. LET $\left.\begin{array}{l} F_B = 200 \text{ LB} \\ M_B = 521 \text{ IN LB} \end{array}\right\}$ CONSIDERS $y_B + \theta = 0$ IN EQ ④ + ⑦

2. $F_T = F_P + F_c + F_H + F_B$

$= 18000 + 156 + 1810 + 200 = 20200$ LB

$M_T = M_P - F_c L_c - M_B - F_B L_B$

$M_T = 800\,000 - 156\times 6.25 - 521 - 200\times 8.75$
$= 797\,000 \text{ IN LB}$

3. $\theta = -\dfrac{20200\times 119.4}{28\times10^6\times 2\times 1556} - \dfrac{797000\times 119.4}{28\times10^6\times 1556} - \dfrac{16\times 3.61\times 119.4^3}{12\times 28\times 10^6\times 1556}$

$= -\dfrac{119.4}{28\times10^6\times1556}\left[\dfrac{20200\times119.4}{2} + 797000 + \dfrac{16\times3.61\times119.4^2}{12}\right]$

$= -.005504 \text{ RAD.}$

$y_H = \dfrac{20200\times119.4^3}{3\times28\times10^6\times1556} + \dfrac{797000\times119.4^2}{2\times28\times10^6\times1556} + \dfrac{16\times3.61\times119.4^4}{16\times28\times10^6\times1556}$

$= \dfrac{119.4^2}{28\times1556\times10^6}\left[\dfrac{20200\times119.4}{3} + \dfrac{797000}{2} + \dfrac{16\times3.61\times119.4^2}{16}\right]$

$= .4103 \text{ IN.}$

$y_B = .4103 + (-.005504)\times 8.75 = .4103 - .0482 = .3621 \text{ IN}$

4. $F_B = 200 - \dfrac{3\times14200\times23.25^2\times1.5}{2\times15.63^3}\left(\overbrace{.3621 - \dfrac{.005504\times15.63}{2}}^{.3191}\right)$

$= 200 - 1440 = -1240 \text{ LB}$

$M_B = 521 - \dfrac{3\times14200\times23.25^2\times1.5}{4\times15.63^2}\left(\overbrace{.3621 - \dfrac{2\times15.63\times.005504}{3}}^{.3048}\right)$

$= 521 - 10800 = -10300 \text{ IN LB}$

2ND TRY

5. LET $F_B = -1240 \text{ LB}$
$M_B = -10300 \text{ LB}$

2. $F_T = 18000 + 1966 - 1240 = 18700 \text{ LB}$

$M_T = 800000 - 975 + 10300 + 1240\times 8.75 = 820000 \text{ IN LB}$

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO. 71983-3

3. $\theta = -\dfrac{119.4}{28\times10^6\times1556}\left[\dfrac{18700\times119.4}{2} + 820000 + 68620\right]$

$= -.005495 \text{ RAD.}$

$y_H = \dfrac{119.4^2}{28\times10^6\times1556}\left[\dfrac{18700\times119.4}{3} + \dfrac{820000}{2} + 51500\right]$

$= .3945 \text{ IN.}$

$y_B = .3945 + (-.005495\times 8.75) = .3945 - .0481 = .3464 \text{ IN.}$

4. $F_B = 200 - 1440\,\dfrac{.3464 - .005495\times\frac{15.63}{2}}{.3191}$

$= 200 - 1369 = -1169 \text{ LB}$

$M_B = 521 - 10800\times\dfrac{.3464 - .005495\times15.63\times2/3}{.3048}$

$= 521 - 10250 = -9730 \text{ IN LB}$

3RD TRY

5. LET $F_B = -1169 \text{ LB}$
$M_B = -9730 \text{ IN LB}$

2. $F_T = 18000 + 1966 - 1169 = 18800 \text{ LB}$

$M_T = 800000 - 975 + 9730 + 1169\times 8.75 = 821000 \text{ IN LB}$

3. $\theta = -\dfrac{119.4}{28\times10^6\times1556}\left[\dfrac{18800\times119.4}{2} + 821000 + 68620\right]$

$= -.005514 \text{ RAD.}$

$y_H = \dfrac{119.4^2}{28\times10^6\times1556}\left[\dfrac{18800\times119.4}{3} + \dfrac{821000}{2} + 51500\right]$

$= .3960 \text{ IN}$

$y_B = .3960 + (-.005514 \times 8.75) = .3478$ IN

5. THIS APPEARS TO BE CLOSE ENOUGH

$F_T = 18800$ LB $\qquad \theta = -.00551$ RAD

$M_T = 821000$ IN LB $\qquad y_B = .348$ IN

BENDING STRESS

$$M_{B_{1,2}} = F_T L_1 + M_T + \frac{w_N'}{2g} a L_1^2$$

$$= 18800 \times 110.4 + 821000 + \frac{16 \times 3.61}{2} \times 110.4^2$$

$$= 3,250,000 \text{ IN. LB.}$$

$$\sigma_b = \frac{M_{B_{1,2}}}{Z} = \frac{3,250,000}{194.5} = \underline{16700 \text{ PSI}}$$

$1.5 S_h = 22500$ PSI (FROM B31.7 FOR CL II)

BELLOWS EQUIV. AXIAL MOVEMENT

$$e = \frac{23.25 \times 1.5}{15.63}\left(-.00551 + \frac{3 \times .348}{15.63}\right) + \frac{110 \times 15.63 \times 3.61}{3 \times 14200 \times 23.25}$$

$$= .136 + .006 = .142 \text{ IN./CONV.}$$

$e_{cyc} = .436$ PRODUCES $N > 7000$ CYCLES

THE ABOVE CALCULATED VALUE IS WELL BELOW THE CYCLIC VALUE USED IN THE LIFE CALCULATIONS. THEREFORE, THE ORIGINAL VALUES ARE ESSENTIALLY JUSTIFIED AS CONSERVATIVE

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO. 71983-3

D. Discussion: As can be observed in the sample calculation above, the resultant bending stress in the penetration is quite low in comparison to allowables. Likewise, the bellows equivalent axial movement per convolution is low (see Section 2). Therefore, the penetration assembly is well able to support the seismic load case as defined by "Ebasco". Furthermore, since the stresses produced by pipe rupture case are low for all penetrations (see Table 5) and considering pipe rupture loads exceed the seismic loads, it would not appear that further seismic calculations are necessary. However, in the interest of conservatism similar calculations were made on Penetrations 5 and 40. The results as expected were quite low. These as well as those from the sample calculation (P36) are presented in the table below:

PEN	σ_b	1.5 S_h	e	e cyc.
5	9610 PSI	22500 PSI	.074 IN.	.303 IN.
36	16700 PSI	22500 PSI	.142 IN.	.436 IN.
40	8020 PSI	22500 PSI	.052 IN.	.455 IN.

Therefore, on the basis of the above results which essentially cover the complete size range of penetrations as well as represent the highest loaded ones, it is now safe to conclude that all of Type III penetrations are suitable for the seismic design loads defined by "Ebasco".

APPENDIX 3G3

REPORT NO. 71983-4

T-766

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO. 71983-4

DESIGN CALCULATIONS
FOR
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT
(FORMERLY HUTCHINSON ISLAND)

TYPE I CONTAINMENT PIPING PENETRATION ASSEMBLIES
P.O. NO. NY-422264

Included are:

Penetration No. 1, Main Steam (SG-1A), I-34MS-28
Penetration No. 2, Main Steam (SG-1B), I-34MS-29
Penetration No. 3, Feedwater, (SG-1A), I-20BF-14
Penetration No. 4, Feedwater, (SG-1B), I-20BF-19

Prepared by: WS Haberman Date: 21 July 1973
W.S. Haberman
Bellows Engineering

Approved by: M. V. Malkmus Date: 7/24/73
M.V. Malkmus
Manager, Bellows Engineering

3G-27

T-766

TUBE TURNS
LOUISVILLE 1, KENTUCKY

TABLE OF CONTENTS

TUBE TURNS
LOUISVILLE 1, KENTUCKY

1. Summary of Report

As substantiated by the calculations herein, all major components have been demonstrated to comply with the design information supplied. The only exception are the flued heads which are discussed in Reports 71983-1 and 71983-6.

TUBE TURNS
LOUISVILLE 1, KENTUCKY

2. Bellows Design Calculations

A. General: The intent of this section is to demonstrate by suitable computation, the structural adequacy of the primary and secondary bellows for the design conditions stated in paragraph B below. The configurations of the bellows are depicted on Drawings 71983-D1.1, C1.2, and C1.3, attached.

B. Design Conditions:

(1) Primary Bellows

Condition	#1(600 Cycles)	#2(7000 Cycles)
Axial Movement	3.68" comp. 1.43 ext.	.5"
Horizontal Movement	± 1.78"	
Vertical Movement	± 1.5"	± .2"

Material - A 240 TP 316L
Pressure - 44 PSIG
Temperature - 264°F
Factory Precompression - 3/4"

(2) Secondary Bellows

Condition	#1(600 Cycles)	#2(7000 Cycles)
Axial movement	±1.43"	.5"
Horizontal movement	±1.78"	
Vertical movement	±1.5"	± .2"

Material - A 240 TP 316L
Pressure - ± 5 PSI
Temperature - 264°F
Factory Precompression - 2-3/8"

C. References:
1. USAS B31.7-1969
2. Standards of the Expansion Joint Manufacturers'Assn., Inc. Third Edition 1969.

D. Bellows Hoop Stress: The bellows hoop or circumferential membrane stress is determined by use of the Barlow formula for Straight pipe suitably modified to account for a convoluted distribution of metal. As applied to a bellows the equation takes the following form:

$$S_b = \frac{Pd}{2tn_p} \frac{\left(\frac{d+w}{d}\right)^{5/3}}{\left(.571 + {}^{2w}/_{q}\right)}$$

WHERE: S_b = Stress (PSI)
P = Design Pressure (PSIG)
d = Bellows Root Diameter (in.)
t = Bellows Wall Thk-1 Ply (in.)
w = Convolution Height (in.)
q = Convolution Pitch (in.)
n_p = No. Bellows Plies

E. Meridional Bending Pressure Stress
The bellows meridional stress (stress tending to crush or collapse the convolutions in an axial direction) is determined by the following theoretical formula, with empirical modifications to correlate with test data.

$$S_m = \frac{Pw^2}{10\ 2/3\ t^2\ n_p}$$

F. Bellows Movement Capacity: The movement capability of a bellows is evaluated on an "equivalent axial movement" per convolution basis. For the secondary seal bellows, the allowable movements have been limited to the following:

Allowable compressive movement (e comp.)

$$= .85\ (q/2 - t\ n_p)$$

Allowable extention movement (e ext.)

$$= .45\ (q/2 - t\ n_p)$$

The allowable for compression is based primarily on physical limitation from the geometry. For extension, the limitation is based on empirical data to prevent dimpling of the convolution crest and to prevent excessively high stresses that would significantly reduce cyclic life.

The axial and lateral movements tabulated in Paragraph B are converted to equivalent axial movement per convolution by the folllwing formulas: (Ref: Standards of the Expansion Joint Manufacturers' Association):

3G-29

T-766

TUBE TURNS
LOUISVILLE 1, KENTUCKY

$e_x = \frac{x}{2n}$ = Axial Traverse per Conv. (+ is compressive)

$e_y = \frac{KDy}{2n(L-C-x/2)}$ = Equivalent Axial Traverse per Conv. (Caused by Lateral Movement)

$e_{pc} = \frac{Xpc}{2n}$ = Axial Traverse due to Precompression (+ is compressive)

$e_{ext} = - e_x - e_{pc} + e_y$ = Total Axial Ext. per Convolution

$e_{comp.} = e_x + e_{pc} + e_y$

= Total axial compression per convolution

WHERE: x = Axial movement (in.)
y = Lateral movement (in.)
x_{pc} = Precompression (in.)
D = Convolution crest dia. (in.)
= $d + 2w$
n = Number of convolutions in one bellows element
L = Overall bellows length
$2C$ = $2nq$
K = Lateral offset constant from EJMA Standards based on L/2C ratio

G. Bellows Spring Rates: The bellows axial spring rate has been proven experimentally to be predicted by the following equation:

$$f_x = 2\ n_p\ (d + w)\ E\ \tau^3/Cu$$

WHERE: f_x = Bellows axial spring force per convolution(LB/In.)
E = Modulus of Elasticity x10^{-6} (PSI) = 27.4
$\tau = \frac{100t}{w}$ = Thickness to height ratio
Cu = Spring rate denominator - an empirical factor which is a function of convolution pitch, height and diameter.
n_p = Number of bellows plies.

The axial spring rate of the total joint can be expressed as:

$$k_x = \frac{F_x}{x}$$

WHERE: $F_x = f_x \; e_x$

$e_x = \frac{x}{n}$

$x = 1$ in.

Therefore by substitution we obtain:

$$k_x = \frac{f_x}{2n} \quad \text{(Lb/In.)}$$

The lateral spring rate can be expressed as:

$$k_y = \frac{F_y}{y}$$

WHERE: $f_y = \frac{f_x D \; e_y}{2L}$

$e_y = \frac{K \; D_y}{2n \; (L - C)}$

$y = 1$ in.

Therefore by substitution we obtain:

$$k_y = \frac{f_x \; D}{2 \; L} \cdot \frac{K \; D}{n(L-C)}$$

$$k_y = \frac{K \; f_x \; D^2}{4 L n \; (L - C)} \quad \text{(LB./IN.)}$$

H. <u>Cyclic Life Determination</u>: It has been proven through extensive testing that the cyclic life of a U-shaped bellows can be predicted by use of the following formula:

$$N = 4\left(\frac{6 S_u}{S_o'}\right)^4$$

WHERE: N = Calculated Number of Cycles

S_u = Ultimate Tensile Strength of Bellows Material (PSI)

$S_o = (1 + P/n_p \, p')^2 \; 675 \, \tau \, E \, e_o / w$

$S_p = 4.5 \, C_u^{4} \, P \, \tau^{2} \, n_p / (1 + P/n_p p')^2$

$S_o' = S_o + S_p$ = Total stress Parameter (PSI)

$P' = S_u t / 2 W_e$

$\tau = \frac{100 t}{w}$ = Thickness to height ratio

$E = 27.4$ = Modulus of Elasticity $\times 10^{-6}$ (PSI)

$e_o = e_{cyc} - e_e/3$ = Effective traverse per convolution (in.)

w = Convolution Height (in.)

S_c = 17,500 PSI = Cold Allowable Bellows Stress

S_H = 16,400 PSI = Hot Allowable Bellows Stress (264°F)

$W_e = W - t/3$ = Effective Conv. Ht. (In.)

n_p = Number of Bellows Plies

$Q = \frac{S_c + S_H}{2 \; S_c}$ = Hot/Cold Stress Range Ratio = .97

$e_e = \frac{W_e^2}{450 t}$ = Elastic Traverse per Convolution (in.)

For purposes of cyclic life determination, the non-cyclic precompressive movement installed at the factory is not considered as part of the total equivalent axial movement per convolution. Therefore, we obtain:

$$e_{cyc} = \left(e_x + e_y\right) \text{max.}$$

TUBE TURNS
DIVISION OF CHEMETRON CORPORATION
LOUISVILLE 1, KENTUCKY

I. Results:

1. The pertinent bellows static pressure stresses are presented in Table 1, page 10. The allowable stresses have been taken from B31.7 Appendix A, Table A.8.

2. The bellows movement capacity evaluation is shown in Table 2 page 11. The allowable equivalent axial traverse per convolution is computed per paragraph F of this report.

3. Computation of axial and lateral spring rates are included in Table 3, page 12.

4. Computation of bellows cyclic life is presented in Table 4, page 13, for the 600 cycle condition. It is obvious that similar computations for the 7000 cycle condition would produce results for in excess of 7000 cycles for all bellows. Therefore, it is redundant to reproduce these herein.

TUBE TURNS
DIVISION OF CHEMETRON CORPORATION
LOUISVILLE 1, KENTUCKY

REPORT NO.

PEN NO.	d (IN)	t (IN.)	w (IN)	q (IN)	n	S_b (PSI)	S_m (PSI)	S_{allow} (PSI)
1, 2 PRIM.	59.3	.05	2.0	1.5	8-8	4260	3300	17500
1, 2 SEC.	69.3				10-10	560	375	
3, 4 PRIM.	44.3				8-8	3240	3300	
3, 4 SEC.	55.3				10-10	450	375	

TABLE 1
BELLOWS STRESS RESULTS

PEN NO.	d (IN)	t (IN)	w (IN)	δ (IN)	n	L (IN)	x_{pc} (IN)	e_{pc} (IN)	e_x (IN)	e_y (IN)	e_{comp} (IN)	e_{allow} (IN)	e_x (IN)	e_y (IN)	e_{ext} (IN)	e_{allow} (IN)
1, 2 PRI	59.3	.05	2.0	1.5	8-8	57	.75	.047	.230	.263	.540	.553	-.089	.248	.290	.293
1, 2 SEC	69.3	↓	↓	↓	10-10	50	2.375	.119	.089	.334	.542	↓	-.089	.321	.291	↓
3, 4 PRI	44.3	↓	↓	↓	8-8	48	.75	.047	.230	.264	.541	↓	-.089	.245	.287	↓
3, 4 SEC	55.3	↓	↓	↓	10-10	45	2.375	.119	.089	.325	.533	↓	-.089	.310	.280	↓

TABLE 2

BELLOWS ELEMENT MOVEMENT CAPACITY

REPORT NO.

TUBE TURNS
LOUISVILLE 1, KENTUCKY

AXIAL AND LATERAL BELLOWS SPRING RATES

PEN. NO.	d (IN)	t (IN)	w (IN)	δ (IN)	n	K	τ	C_u	f_x (LB/IN/conv)	L (IN.)	L-C (IN)	k_x (LB/IN.)	k_y (LB/IN.)
1, 2 PRI	59.3	.050	2.0	1.5	8-8	1.23	2.50	2.77	37900	57	45	2370	2280
1, 2 SEC	69.3	↓	↓	↓	10-10	1.34	2.50	2.79	43800	50	35	2190	4500
3, 4 PRI	44.3	↓	↓	↓	8-8	1.28	2.50	2.74	28900	48	36	1810	1560
3, 4 SEC	55.3	↓	↓	↓	10-10	1.38	2.50	2.76	35600	45	30	1780	3200

TABLE 3

REPORT NO.

TUBE TURNS
LOUISVILLE 1, KENTUCKY

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO.

BELLOWS CYCLIC LIFE RESULTS

PEN NO.	d (IN.)	t (IN.)	w (IN.)	g (IN.)	R	C_d	e_x (IN.)	e_y (IN.)	e_{cyc} (IN.)	S_o (PSI)	S_p (PSI)	S_o' (PSI)	N CALC	N REQD
1,2 PXI	59.3	.05	20	1.5	8-8	2.77	.230	.263	.493	11000	910	11900	4240	600
1,2 SEC	69.3				10-10	2.79	.089	.334	.423	9050	109	9160	>7000	600
3,4 PXI	44.3				8-8	2.74	.230	.264	.494	11000	862	11900	4240	600
3,4 SEC	55.3				10-10	2.76	.089	.325	.414	8840	104	8940	>7000	600

TABLE 4

T-766

TUBE TURNS
LOUISVILLE 1, KENTUCKY

3. Process Pipe Wall Thickness Check

A. General: The process pipe minimum wall requirements for design pressure and temperature as specified on the "Ebasco" drawings has been computed per the following equation from B31.7:

$$t_m = \frac{PD}{2\,(S^E + .4P)}$$

The S^E values in the above equation were taken from B31.7 Appendix A, Table A8. The requried nominal thickness was taken as:

$$t_{NOM} = t_m/.875 \quad \text{(For SMLS Pipe)}$$

B. References:
1. USAS B31.7 - 1969
2. Ebasco Drawing No. 8770-6-213

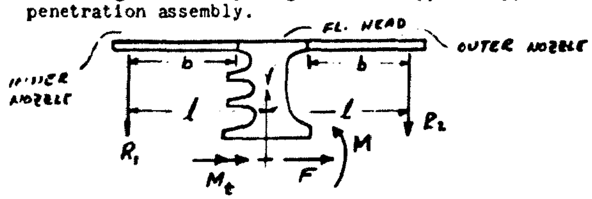
C. Results:
The results of this evaluation are presented in Table 5 below. Note that all process pipe possess sufficient wall thickness to resist the internal pressures specified.

PEN	P	D	SE	tm	t_{NOM}	Act. t_{NOM}
1,2	985	34	17500	.936	-	1.125 Min.
3,4	1100	20	15000	.712	.814	1.031

TABLE 5

4. <u>Secondary Seal Nozzle Stresses Due to Pipe Rupture Loading Condition:</u>

A. <u>General</u>: For purposes of analysis, consider the following "free body diagram" of a typical type I penetration assembly.



In the above M,F,V and M_t each represent a combined effect of thermal expansion and pipe rupture loading. This is arrived at by assuming one process pipe hub to undergoing its normal piping flexibility thermal expansion reactions while the other hub is loaded by pipe rupture reactions. For conservatism, these are assumed additive. For analysis purposes, the following load combinations should be considered as to their relative effects:

1. Bending and Lateral (M,V)
2. Torsion and Lateral(M_t, V)
3. Bending and Axial (M,F)

Since, the load transmitting structural attachments consist of loose rings at the ends of the nozzles and trunnions on the flued heads, it becomes obvious the worst condition from a nozzle standpoint is the Bending and Lateral case. This is due to the trunnion resistance of the axial and torsional loads. Therefore, from statics we have:

$$R_1 + R_2 = V$$

$$-R_1 \ell + R_2 \ell = M$$

Solving we obtain:

$$R_2 = \frac{M}{2\ell} + \frac{V}{2}$$

$$R_1 = \frac{-M}{2\ell} + \frac{V}{2}$$

Converting to a max nozzle bending moment we obtain:

$$(M_b)_{max.} = b \quad R_2$$

Therefore, the maximum primary bending stress becomes:

$$\sigma_b = \frac{M_b}{Z}$$

Where: σ_b = Maximum bending stress (PSI)

M_b = Maximum bending moment (in. lb.)

Z = Nozzle section Modulus (in.^{3})

The above constitutes a maximum primary bending stress and is therefore comparable to an assumed allowable of pipe yield per Section III. However, since the loads are transmitted from the nozzles to loose ring flanges, it becomes necessary to determine the discontinuity bending effects imposed (stress causing nozzle to deform at the ends). This effect could result in an ovalization of the ring end sufficient to collapse or yield a cylinder of insufficient thickness. To arrive at a suitable stress equation to evaluate this effect, it was necessary to employ strain energy techniques. In the interest of brievity only the highlights of the development will be presented here.

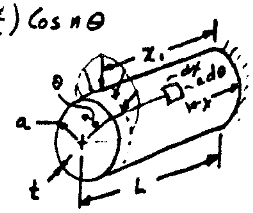
Basically, it was assumed the nozzle can be represented as a cylinder fixed at one end and free at the other. A shear load is applied near the free end distributed over half the circumference as a cosine function. The potential energy (V) of the applied shear distribution is equated to the strain energy (U) of the cylinder using the "First Minimal Principle" or the minimum of total potential:

$$\frac{\partial U}{\partial q_i} + \frac{\partial V}{\partial q_i} = 0$$

Where q_i is any arbitrary variable or parameter.

For our deflection equation it was assumed:

$$w = \sum C_n \left(1 - \cos \frac{\pi x}{2L}\right) \cos n\theta$$



The strain energy of the cylinder is represented by:

$$U = \frac{D}{2} \int_0^L \int_0^{2\pi} \left[\chi_x^2 + \chi_\theta^2 + 2\mu \chi_x \chi_\theta + 2(1-\mu) \chi_{x\theta}^2 \right] a \, d\theta \, dx$$

Where:

$X_x = \frac{\partial^2 w}{\partial x^2}$ = meridional bending curvature

$X_\theta = \frac{1}{a^2} \frac{\partial^2 w}{\partial \theta^2}$ = tangential bending curvature

$X_{x\theta} = \frac{1}{a} \frac{\partial^2 w}{\partial \theta \partial x}$ = torsional bending curvature

The applied shear load is assumed as:

$$q_\theta = \frac{2R}{a\pi} \cos\theta$$

Therefore the potential of the applied load becomes:

$$-V = 2 \int_0^{\pi/2} q_\theta \, w \cdot a \, d\theta$$

Derivatives of U and V are taken and equated as follows:

$$\frac{\partial U}{\partial C_m} + \frac{\partial V}{\partial C_m} = 0$$

The above equation is then solved for the unknown C_m which represents the mth term of the series $\sum C_n \left(1 - \cos \frac{\pi x}{2L}\right) \cos n\theta$ and hence an expression defining all terms of the series is now available. The final deflection equation becomes:

$$w = \frac{-8a^3 R}{\pi D} \left(1 - \cos \frac{\pi X_1}{2L}\right) \sum_{n=2,4,6\ldots}^{\infty} \frac{(-1)^{n/2} \left(1 - \cos \frac{\pi X}{2L}\right) \cos n\theta}{(n^2-1) \left\{ \frac{\pi}{16} \left(\frac{a}{L}\right)^3 + \frac{n^4}{a/L} (3\pi - 8) + \frac{\pi^2 n^2}{2} (\pi - 2\mu) \frac{a}{L} \right\}}$$

Where: $D = \frac{Et^3}{12(1-\mu^2)}$

The circumferential bending stress equation is:

$$\sigma_{\theta b} = \frac{-Et}{2(1-\mu^2)} (\chi_\theta + \mu \chi_x)$$

Therefore, assuming μ = .3 and X = L, we obtain by substitution:

$$\sigma_{\theta b} = \frac{15.3R}{t^2} \left(1 - \cos \frac{\pi X_1}{2L}\right) \sum_{n=2,4,6\ldots}^{\infty} \frac{n^2 (-1)^{n/2} \cos n\theta}{(n^2-1) \left\{ 19.13 \left(\frac{a}{L}\right)^3 + \frac{1.425 n^4}{a/L} + 9.581 n^2 \left(\frac{a}{L}\right) \right\}}$$

Examination of the above leads to the conclusion that the maximum stress occurs at θ = 90°. This is due to the fact that all terms of the series become positive. Therefore, the final equation becomes:

$$\sigma_{\theta b} = \frac{15.3R}{t^2} \left(1 - \cos \frac{\pi X_1}{2L}\right) \sum_{n=2,4,6\ldots}^{\infty} \frac{\frac{n^2}{n^2-1}}{19.13 \left(\frac{a}{L}\right)^3 + \frac{1.425}{a/L} + 9.581 n^2 \left(\frac{a}{L}\right)}$$

Where:
- R = Maximum transverse load (lb.)
- a = Radius of Nozzle (in.)
- t = Thickness of nozzle (in.)
- L = Length of nozzle (in.)
- X_1 = Distance of load plane from fixed end (in.)

B. References:

1. USAS B31.7 - 1969
2. Ebasco informal memo dated December 22, 1971 L.J. Sas to W.S. Haberman
3. "Advanced Strength of Materials" by J.D. Dan Hartog
4. "Theory of Plates and Shells" by S. Timoshenko

C. Computations

1. Main Steam Penetrations (P1 + P2)

M = 49,200,000 IN LB	b = 38.25"
V = 191000 LB	a = 33.8 in.
λ = 44 IN	t = 2.5 in.
L = 44.5 IN	X_1 = 40.75

$$P_2 = \frac{49200000}{2 \times 44} + \frac{191000}{2} = 655000 \text{ LB}$$

$$M_b = bP_2 = 38.25 \times 655000 = 25{,}100{,}000 \text{ IN LB}$$

Primary Bending Stress:

$$\sigma_b = \frac{M_b}{Z} = \frac{25{,}100{,}000}{\pi \times 33.8^2 \times 2.5} = \underline{2740 \text{ PSI}}$$

Local Discontinuity

a/L = .760

$$x_1/L = \frac{40.75}{44.5} = .916 \qquad \frac{\pi x_1}{2L} = .916 \times 90 = 82.4°$$

$$\sigma_t = \frac{15.3 \times 655000}{} (1 - \cos 82.4°) \times .0221$$

$$= \underline{30100 \text{ PSI}}$$

2. Feedwater (P3 & P4)

M = 5,040,000 IN LB.	b = 39.5
V = 86,000 LB	a = 27.6 IN
λ = 44 IN.	t = .1 IN
L = 44 IN.	X_1 = 41.5 IN.

$$R_2 = \frac{5040000}{2 \times 44} + \frac{86000}{2} = 100{,}000 \text{ LB}$$

$$M_b = b \times R_2 = 39.5 \times 100000 = 3{,}950{,}000$$

Primary Bending Stress

$$\sigma_b = \frac{M_b}{Z} = \frac{3950000}{\pi \times 27.6^2 \times .1} = \underline{1650 \text{ PSI}}$$

$$a/L = \frac{27.6}{44} = .627$$

$$x_1/L = \frac{41.5}{44} = .943 \qquad \frac{\pi x_1}{2L} = .943 \times 90 = 84.9°$$

Local Discontinuity

$$\sigma_{\theta b} = \frac{15.3 \times 100000}{1} (1 - \quad 84.9°) \times .0224$$

$$= \underline{31200 \text{ PSI}}$$

D. Results:

As can easily be observed all values are within allowable taken as 1.5 Sm for the primary stresses (σ_b) and 3 Sm for the secondary stresses, ($\sigma_{\theta b}$). Sm = 23300 PSI @ 100°F for A 515 Gr. 70

5. Lateral Jet Force Calculation

A. General: The jet forces for a longitudinal break are determined per the "Ebasco" spec. as follows:

$F_j = KPA$

WHERE: F_j = Jet Force (LB.)
K = Phase Factor (pg. 9 of spec.)
P = Operating pressure (psig)
A = Pipe Flow Area (in.^{2})

Per the "Ebasco" Specification.

Break Length (ℓ) = 2 x Pipe Dia. (D)

Break Area = A_f

Therefore, Break Width (c) = $A_f/2D$ = A_f/ℓ

B. References:
1. Ebasco Specification No. FLO-8770.124
2. Ebasco Dwg. No. 8770-G-213

C. Results: The results of this computation using operating pressures and piping dimensions from Ebasco Dwg. No. 8770-G-213 are presented in Table 6 below. The forces will be used in the subsequent section.

PEN. NO	(IN)	t (IN.)	A_t (IN^2)	P (PSIG)	K	L (IN)	C (IN)	F_j (LB)
1,2	34	1.25	779	885	.63	68	11.5	868000
3,4	20	1.031	253	1050	.97	40	6.33	516000

TABLE 6

Per the Ebasco specification; the above K values double in the case of fluid rebound. Hence, the forces computed above reflect this effect.

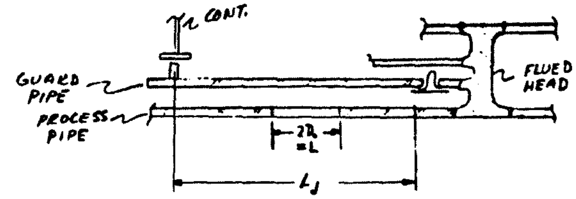
2-766

TUBE TURNS
LOUISVILLE 1, KENTUCKY

6. Guard Pipe Stresses due to Jet Impingement

A. General: To determine the effect of a slot break within the penetration assembly, a conservative approach will be taken as follows:

1. The maximum bending stress that could occur in a guard pipe with a hinge would be at the center of the assembly as shown below:



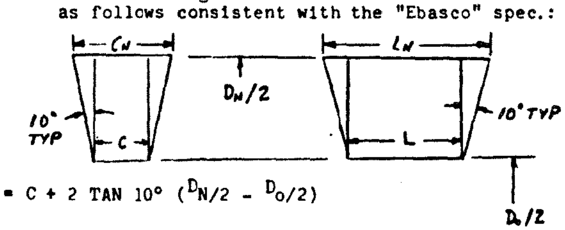
$$M_{max} = \frac{2F_j}{4}(L_j - L/2) = \frac{F_j}{2}(L_j - L/2)$$

$$\sigma_{bx} = M_{max}/Z$$

WHERE $Z = \pi\left(\frac{D_N + t}{2}\right)^2 t$

Note: The values of F_j are taken from Section 5 of this report.

2. The second area of concern is the hoop stress caused in the nozzle or guard pipe due to the jet force striking a finite area. The area is developed as follows consistent with the "Ebasco" spec.:



$C_N = C + 2 \text{ TAN } 10° \,(D_N/2 - D_O/2)$

$= C + (D_N - D_O) \text{ TAN } 10°$

$L_N = L + (D_N - D_O) \text{ TAN } 10°$

$A = C_N L_N$

$= [C + (D_N - D_O) \text{ TAN} 10°]\,[L + (D_N - D_O) \text{ TAN } 10°]$

$$P = \frac{F_J}{A}$$

$$\sigma_t = \frac{P(D_N + t)}{2t}$$

WHERE: D_N = G.P. I.D. (in.)

D_o = Process Pipe (O.D.)

t = Nozzle thickness (in.)

F_J = Jet Force (LB.)

c = Slot Width (in.)

L = Slot length (in.)

B. References:
1. Ebasco Spec. No. FLO-8770-124
2. "Formulas for Stress and Strain" by R.J. Roark
3. USAS B31.7-1969

C. Results: The tabulated results of the above computations are presented in Tables 7 and 8 below. This condition has been assumed as faulted thereby justifying a specified minimum yield allowable stress.

PEN. NO.	F_J (LB.)	L (IN)	L_J (IN)	(IN^3)	(PSI)	S_y (PSI)
1	868000	68	166	2440	11740	29400
2	868000	68	168	2440	11920	29400
3	516000	40	188	923	23400	30800
4	516000	40	177	923	21800	30800

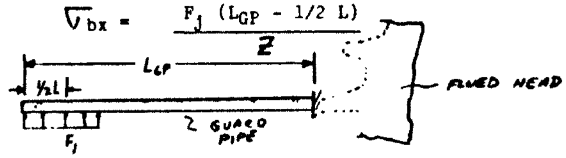
TABLE 7

PEN. NO.	D_N (IN)	t (IN.)	D_o (IN)	C (IN)	A_2 (IN^2)	t (PSI)	S_y (PSI)
1	44	1.5	34	11.5	925	14200	29400
2	44	1.5	34	11.5	925	14200	29400
3	29.5	1.25	20	6.33	334	19000	30800
4	29.5	1.25	20	6.33	334	19000	30800

TABLE 8

However, if the hinge were not present the bending stresses in the guard pipe adjacent to the flued head would be represented by the following formula:

$$\sigma_{bx} = \frac{F_J\,(L_{GP} - 1/2\ L)}{Z}$$

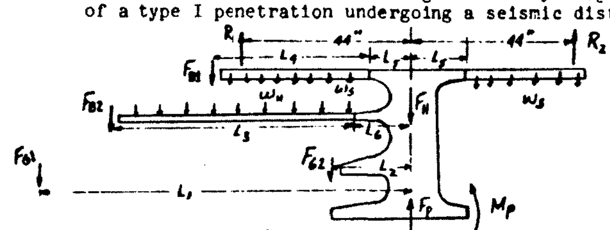


The results are shown in the table below thus demonstrating the imposition of stresses in excess of yield if a hinge is not employed.

PEN. NO.	L_{GP}	1/2 L	σ_{by}	σ_y (PSI)
1	195	34	57200	27800
2	200	34	59000	27800
3	225	20	114000	29100
4	213	20	107000	29100

B. Penetration Seismic Analysis

A. General: Consider the following "free body diagram" of a type I penetration undergoing a seismic disturbance



Where: $F_{B_1} = \frac{a}{g}(W_{B_1} + \frac{1}{2} W_{CS_1})$ =bellows seismic force (lb.)

W_{B_1} = wgt. of one bellows element (LB.)

W_{CS_1} = wgt. of bellows center spool (LB.)

F_{B_2} = a/g (W_{B2} + ½ W_{CS2}) = bellows seismic force (lb.)

W_B = wgt. of one bellows element (LB.)

W_{CS}= wgt. of bellows center spool (LB.)

$F_{G1} = F_{G2}$ = 1/2 a/g W_{GP}

W_{GP}= Wgt. of gurad pipe (LB.).

F_H = a/g W_H

W_H = wgt. flued head (LB.)

TUBE TURNS
LOUISVILLE 1, KENTUCKY

W_s = a/g W_s' = Sec. Seal nozzle seismic load (LB/in)

W_s' = Pipe wgt. (LB/in)

W_N = a/g W_N' = Cont. nozzle seismic load (LB/in)

W_N' = Cont. nozzle pipe wgt. (LB/in)

R_1, R_2 = Anchor structure load bearing reactions (LB)

Fp = D.B.E. + Thermal Expansion Piping Load (LB)

Mp = D.B.E.+thermal expansion piping load (IN LB)

Using the principles of statics we obtain:

$$R_1 + R_2 = F_T - F_p$$

$$44\ (R_2 - R_1) = -M_p - M_T$$

WHERE:

$$M_T = -42\ W_s\ (L5+21) + W_s L4\ (L5 + 1/2\ L4) + W_N\ L3\ (L6 + 1/2 L3) + F_{B1}\ (L4 + L5) + F_{B2}\ (L3 + L6) + F_{G1} L_1 + F_{G2} L_2$$

$$F_T = F_H + W_s\ (42 + L_4) + W_N L3 + F_{B1} + F_{B2} + F_{G1} + F_{G2}$$

Solving we obtain:

$$R_1 = \frac{F_T - F_P}{2} + \frac{M_P + M_T}{88}$$

$$R_2 = \frac{F_T - F_P}{2} - \frac{M_P + M_T}{88}$$

TUBE TURNS
LOUISVILLE 1, KENTUCKY

The maximum nozzle bending moment on the bellows side becomes:

$$M_{b1} = R_1\ (44 - L_5) - F_{B1} L_4 - W_s L_4^2/2$$

The maximum nozzle bending moment on the outboard end becomes:

$$M_{b2} = R_2\ (44 - L_5) - 882\ W_s$$

The maximum primary bending stress becomes:

$$\sigma_b = \frac{(M_b)_{max}}{Z}$$

The maximum secondary stress at the end of the nozzle is determined by using the equation developed in section 4 of this report.

B. References-

1. Ebasco Specification No. FLO-8770-124
2. Ebasco Informal memo dated December 22, 1971 L.J. Sas to W.S. Haberman
3. ASME Section III
4. USAS B31.7-1969

C. Computations:

1. Main Steam (P2)
For the purposes of this computation the following is assumed:

W_{B1}	= 320 LB.	W_{GP}	= 13600	L_1	= 208"
W_{CS1}	= 730 LB.	W_{FH}	= 8500 LB	L_2	= 17"
W_{B2}	= 220 LB.	W_s'	= 160 LB/IN	L_3	= 102"
W_{CS2}	= 950 LB.	W_N'	= 28 LB/IN	L_4	= 42.25"
				L_5	= 5.75"
				L_6	= 6.75"

$F_p = V_{TH} + V_{DBE}$ = 120000 LB.

$M_p = 2(M_{TH} + M_{DBE})$ = 2 x 30000000=60,000,000 IN.LB.

The above assumes the moments from both inboard and outboard are additive while shear from both sides negate each other. Therefore, for conservatism, the shear on the left side will be neglected.

$a = \sqrt{(2g)^2 + (2g)^2} = 3.61g$

$F_{B1} = 3.61\ (320 + 1/2 \times 730) = 2470$ lb.

$F_{B2} = 3.61\ (220 + 1/2 \times 950) = 2510$ lb.

$F_{G1} = F_{G2} = 1/2 \times 3.61 \times 13600 = 24500$ lb.

$F_H = 3.61 \times 8500 = 30700$ lb.

$Ws = 3.61 \times 160 = 578$ lb/in.

$W_N = 3.61 \times 28 = 101$ lb./in.

$M_T = -42 \times 578\ (5.75 + 21) + 578 \times 42.25\ (5.75 + 21.12)$

$+ 101 \times 102\ (6.75 + 51) + 2470\ (42.25 + 5.75)$

$+ 2510\ (102 + 6.75) + 24500\ (208 + 17) = 6{,}500{,}000$ lb.

$F_T = 30700 + 578\ (42 + 42.25) + 101 \times 102$

$+ 2470 + 2510 + 2 \times 24500 = 144{,}000$ lb.

We must now consider two cases. One where M_p & F_p are positive and a second where they are negative.

CASE 1 (M_p & F_p Positive)

$R_1 = \dfrac{144000 - 120000}{2} + \dfrac{60{,}000{,}000 + 6{,}500{,}000}{88}$

$= 12000 + 756000 = \underline{768000 \text{ lb.}}$

$R_2 = \dfrac{144000 - 120000}{2} - \dfrac{60{,}000{,}000 + 6{,}500{,}000}{88}$

$= 12000 - 756000 = \underline{-742000 \text{ lb.}}$

Case 2 ($M_p + F_p$ both negative)

$R_1 = \dfrac{144000 + 120000}{2} + \dfrac{-60{,}000{,}000 + 6{,}500{,}000}{88}$

$= 132{,}000 - 608{,}000 = \underline{-476{,}000 \text{ LB.}}$

$R_2 = \dfrac{144000 + 120000}{2} \quad \dfrac{-60{,}000{,}000 + 6{,}500{,}000}{88}$

$= 132{,}000 + 608{,}000 = \underline{740{,}000 \text{ LB.}}$

Inspection of the above reveals the maximum value of R_1 = 768,000 lb. while the maximum value of R_2 = -742,000 LB.

$M_{b1} = 768000\ (44-5.75) - 2470 \times 42.25 - 578 \times 42.25^2/2$

$= 28{,}800{,}000$ IN LB

$M_{b2} = -742000\ (44-5.75) - 882 \times 578$

$= -28{,}900{,}000$ IN LB.

Therefore the maximum primary bending stress becomes:

$$\sigma_b = \frac{M_{b1}}{Z} = \frac{-28{,}900{,}000}{\pi \times 33.8^2 \times 2.5} = \underline{-3220 \text{ PSI}}$$

The maximum secondary stress becomes:

$$\sigma_{ob} = \frac{768000}{655000} \times 30100 = \underline{35300 \text{ PSI}}$$

from Section 4

2. Feedwater (P3)
Assume the following:

W_{B1} = 250 LB.	W_{GP} = 8600 LB.	L_1 = 230 IN.
W_{CS1} = 460 LB.	W_{FH} = 3500 LB.	L_2 = 16 IN.
W_{B2} = 160 LB.	Ws' = 52 LB/IN	L_3 = 130 IN.
W_{CS2} = 520 lb.	W_N' = 21 LB/IN	L_4 = 55.4 IN.
		L_5 = 4.5 IN.
		L_6 = 5.5 IN.

$F_P = V_{TH} + V_{DBE} = 98000$ LB.

$M_P = 2\,(M_{TH} + M_{DBE}) = 10{,}100{,}000$ IN LB

$a = 3.61g$

$F_{B1} = 3.61\,(250 + 1/2 \times 460) = 1730$ LB

$F_{B2} = 3.61\,(160 + 1/2 \times 520) = 1520$ LB.

$F_{G1} = F_{G2} = 1/2 \times 3.61 \times 3600 = 15500$ LB.

$F_H = 3.61 \times 3500 = 12600$ LB.

$W_S = 3.61 \times 52 = 188$ LB/IN

$W_H = 3.61 \times 21 = 76$ LB/IN

$$
\begin{aligned}
M_T = &-42 \times 188\,(4.5 + 21) + 188 \times 55.4\,(4.5 + 27.7) \\
&+ 76 \times 130\,(5.5 + 65) + 1730\,(55.4 + 4.5) \\
&+ 1520\,(130 + 5.5) + 15500\,(230 + 16) \\
= &\ 4{,}950{,}000 \text{ IN LB.}
\end{aligned}
$$

$$
\begin{aligned}
F_T = &\ 12600 + 188\,(42 + 55.4) + 76 \times 130 + 1520 + 1730 \\
&+ 2 \times 15500 \\
= &\ 75000 \text{ LB.}
\end{aligned}
$$

Case 1 ($M_P + F_P$ Positive)

$$R_1 = \frac{75000 - 98000}{2} + \frac{10{,}100{,}000 + 4{,}950{,}000}{88}$$

$$= -11500 + 171{,}000 = \underline{160{,}000 \text{ LB.}}$$

$$R_2 = \frac{75000 - 98000}{2} - \frac{10{,}100{,}000 + 4{,}950{,}000}{88}$$

$$= -11500 - 171{,}000 = 183{,}000 \text{ lb.}$$

T-766

TUBE TURNS
LOUISVILLE 1, KENTUCKY

Case 2 ($M_P + F_P$ negative)

$$R_1 = \frac{75000 + 98000}{2} + \frac{-10{,}100{,}000 + 4{,}950{,}000}{88}$$

$$= 86500 - 58500 = \underline{28000 \text{ LB.}}$$

$$R_2 = \frac{75000 + 98000}{2} - \frac{-10{,}100{,}000 + 4{,}950{,}000}{88}$$

$$= 86500 + 58500 = \underline{145{,}000 \text{ LB.}}$$

TUBE TURNS
LOUISVILLE 1, KENTUCKY

Inspection of the above values indicate the maximum moment will be produced by $R_2 = -183000$

$$M_{b_2} = -183000\ (44 - 4.5) - 882 \times 188$$
$$= -7{,}390{,}000 \text{ IN LB.}$$

Therefore the maximum primary bending stress becomes:

$$\sigma_b = \frac{M_{b_2}}{Z} = \frac{-7{,}390{,}000}{\pi \times 27.5^2 \times 1} = -3110 \text{ PSI}$$

The maximum secondary stress becomes:

$$\sigma_{\theta b} = \frac{-183{,}000}{100{,}000} \times 31200 = -57100 \text{ PSI}$$

D. Results:

It is easily seen that the above stress values fall within allowable taken as 1.5 S_m for primary stresses and 3 S_m taken for secondary stresses. S_m = 233000 PSI for A516 Gr. 70 plate at 100°F.

APPENDIX D3G4

REPORT NO. 71983-5

TUBE TURNS
LOUISVILLE 1, KENTUCKY

REPORT NO.: 71983-5

FLUED HEAD

STRESS REPORT

FOR

FLORIDA POWER & LIGHT COMPANY

HUTCHINSON ISLAND PLANT

CONTAINMENT PIPING PENETRATION ASSEMBLIES

P.O. NO. NY-422264

3G-44

Included are:

Penetration No. 5 - Blowdown (SG-1A), I-2B-1
Penetration No. 26- Letdown Line, I-2CH-103
Penetration No. 36- Safety Injection Loop 1A2, I-6 SI-113
Penetration No. 40- Shutdown Cooling, I-10 SI-422

PREPARED BY: W.S. Haberman — W.S. Haberman, PE, KY 7592 — DATE: 31 Jan 73

APPROVED BY: M.V. Malkmus — M.V. Malkmus — DATE: 1/31/73

Although this stress report is believed to be accurate, nothing contained herein shall be construed as establishing any warranty, express or implied.

TUBE TURNS
LOUISVILLE 1, KENTUCKY

Report No.71983-5

TABLE OF CONTENTS

TUBE TURNS
LOUISVILLE 1, KENTUCKY

Section III - Penetration 26 Computer Results

18. Pressure, Thermal, and DBE (Transverse & Torsion)

Section IV - Penetration 36 Computer Results

1. Material Property Data
2. Thermal Model Plot
3. Steady State Thermal Data
4. Transient Thermal Data
5. Stress Model Plot
6. Stress Analysis Input Table
7. Internal Pressure Stresses
8. Thermal Gradient Stresses
9. Transient Thermal Stresses
10. Axial Load Stresses
11. Torsional Load Stresses
12. Transverse Shear Stresses
13. Bending Moment Stresses
14. Bending Moment (Alt. End) Stresses
15. Normal Operating (Axial & Bending)
16. Normal Operating (Transverse and Bending)
17. Normal Operating (Transverse and Torsion)
18. Pressure, Thermal, and DBE (Axial and Bending)
19. Pressure, Thermal, and DBE (Transverse and Bending)
20. Pressure, Thermal, and DBE (Transverse and Torsion)
21. Transient Combination 1 (.5 x Trans. Therm. Stress = Alt. Stress)
22. Transient Combination 2 (.5 x Trans. Therm. Stress - .5 x S.S. Thermal Gradient Stress)
23. Transient Combination 3 (.5 x Thermal Gradient Stress)

Section V - Penetration 40 Computer Results

1. Material Property Data
2. Thermal Model Plot
3. Thermal Results
4. Stress Model Plot
5. Stress Analysis Input Table
6. Internal Pressure Stresses
7. Thermal Gradient Stresses
8. Axial Load Stresses
9. Torsional Load Stresses
10. Bending Moment (Alt. End) Stresses
11. Transverse Shear Stresses
12. Bending Moment Stresses
13. Normal Operating (Axial & Bending)
14. Normal Operating (Transverse and Bending)
15. Normal Operating (Transverse and Torsion)
16. Pressure, Thermal and DBE (Axial and Bending)
17. Pressure, Thermal and DBE (Transverse and Bending)
18. Pressure, Thermal and DBE (Transverse and Torsion)

TUBE TURNS
LOUISVILLE 1, KENTUCKY

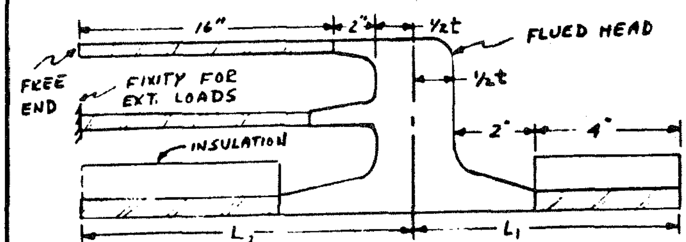
REPORT SUMMARY

Introduction

The following, when combined with the balance of items listed in the index, constitutes the stress report on four (4) flued head forgings to be supplied as part of the containment penetration piping assemblies for the Hutchinson Island Plant. All calculations for stress distributions were made using finite element computer techniques. Description data relative to the program and its verification is attached in Section I-3 and I-4.

Basic Approach

For analysis purposes the flued head and attached piping model was assumed as shown below:



PEN	1/2t	t	L2	L1	OPERATING PRESS.	OPERATING TEMP.	THK. INSUL.
5	1	2	19	7	885 PSIG	520°F	2
26	1-1/2	3	19-1/2	7-1/2	2200 PSIG	450°F	1-1/2
36	1-3/4	3-1/2	19-3/4	7-3/4	2235 PSIG	330°F	2
40	1-1/2	3	19-1/2	7-1/2	300 PSIG	300°F	2

The table below combines, Thermal Expansion and Operational basis earthquake loads per Ebasco Memo, December 22, 1972.

PEN	(F) AXIAL	(M) BENDING	(V) TRANS	(Mt) TORSION
5	1650 lb.	16500 in. lb.	1650 lb.	16500 in. lb.
26	300	2250	300	2250
36	13500	200000	13500	300000
40	9150	474000	9150	474000

If we consider Thermal Expansion plus design basis Earthquake we have the following:

PEN	AXIAL	BENDING	TRANS	TORSION
5	2200 lb.	22000 in. lb.	2200 lb.	22000 in. lb.
26	400	3000	300	3000
36	18000	400000	18000	400000
40	12600	632000	12600	632000

The ratio between the above two sets of loads becomes 1.333 considering operating basis earthquake to be one half of the design basis values. Further, the above values act at the center of the flued head and comprise pipe loads applicable to both the inboard and outboard side in the worst combination from an overall head stress standpoint. For the actual stress runs the above loads were converted to statically equivalent loads applied to the ends of the stress model shown in the above sketch. Combinations produced in the combined stress results runs include the following for both the normal operating conditions (includes pressure, thermal, and operating basis earthquake) and the emergency conditions (includes pressure, thermal, and design basis earthquake)

Combination 1

1. Process pipe pressure stresses
2. Head Thermal gradient stresses
3. Axial thermal load stresses
4. Bending thermal load stresses
5. Axial seismic load stresses
6. Bending seismic load stresses

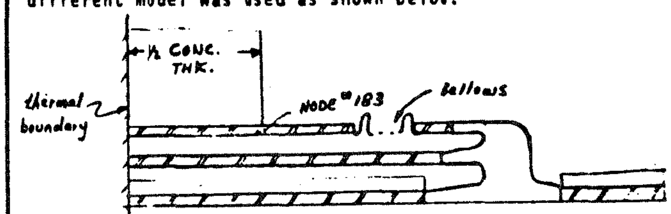
Combination 2

1. Process pipe pressure stresses
2. Head thermal gradient stresses
3. Transverse thermal load stresses
4. Bending thermal load stresses
5. Transverse seismic load stresses
6. Bending seismic load stresses

Combination 3

1. Process pipe pressure stresses
2. Head thermal gradient stresses
3. Transverse thermal load stresses
4. Torsional thermal load stresses
5. Transverse seismic load stresses
6. Torsional seismic load stresses

In addition to the above finite element stress analysis for the flued heads, the axial thermal gradients were investigated on Penetration 26 to demonstrate in imposed concrete shield vessel temperature of 150°F or less. For this analysis, a slightly different model was used as shown below:



Finally the pipe rupture loading contions have been treated in Report No. 71983-1 previously submitted. Likewise, the balance of penetration major components have been evaluated in Report 71983-3.

TUBE TURNS
LOUISVILLE 1, KENTUCKY

RESULTS SUMMARY

The results for the four (4) heads analyzed are presented below:

PENETRATION NUMBER 5

Combination	Max Stress Intensity(PSI)	Stress Location	Allowable Stress
#1 Normal OP. (Ax+BEN)	23776	OD Outboard Process Pipe Hub (#64)	57729
#2 Normal OP. (Trans+BEN)	19845	OD Inboard Process Pipe Hub (#77)	57124
#3 Normal OP. (Trans+Tor)	24575	OD Outboard Process Pipe Hub (#64)	57729
#4 EMER. (Ax-BEN)	31158	OD Outboard Process Pipe Hub (#64)	57729
#5 EMER. (Trans+BEN)	26940	CD Inboard Process Pipe Hub (#77)	57124
#6 EMER (Trans +Tor)	35689	CD Outboard Process Pipe Hub (#64)	57729

PENETRATION NUMBER 26

Combination	Max Stress Intensity(PSI)	Stress Location	Allowable Stress
#1 Normal OP.(Ax+BEN)	25588	Inboard Process Pipe Fillet RAD(#183)	54188
#2 Normal OP.(Trans+BEN)	25608	"	"
#3 Normal OP.(Trans+Tor)	25616	"	"
#4 EMER. (Ax+BEN)	25581	"	"
#5 EMER.(Trans+BEN)	25608	"	"
#6 EMER.(Trans+Tor)	25620	"	"

3G-47

TUBE TURNS
LOUISVILLE 1, KENTUCKY

PENETRATION NUMBER 36

Combination	Max Stress Intensity	Stress Location	Allowable Stress
#1 Normal OP.(Ax+Ben)	19036	ID Head(#125)	57852
#2 Normal Op.(Trans+BEN)	19118	ID Head(#125)	57852
#3 Normal OP.(Trans+Tor)	20402	Outboatd Process Pipe Hub (#23)	57744
#4 EMER. (Ax+BEN)	23569	OD Inboard Process Pipe Hub (#91)	57508
#5 EMER.(Trans+BEN)	22365	OD Inboard Process Pipe Hub(#91)	57508
#6 EMER.(Trans+Tor)	26775	Outboard Process Pipe Hub (#23)	57744

Max Transient Stress = 25430 (ALT) @505 cycles /Usage Factor < .01

Stress located at I.D. of Outboard Process Pipe Hub

PENETRATION NUMBER 40

Combination	Max Stress Intensity	Stress Location	Allowable Stress
#1 Normal OP.(Ax+BEN)	22542	ID Inboard Process Pipe Hub (#86)	56100
#2 Normal OP.(Trans+BEN)	22300	OD Outboard Process Pipe Hub (#22)	"
#3 Normal OP.(Trans+Tor)	22439	"	"
#4 EMER. (Ax+BEN)	25945	"	"
#5 EMER. (Trans+BEN)	25918	"	"
#6 EMER. (Trans+Tor)	29889	"	"

PENETRATION 26 AXIAL THERMAL GRADIENT

The results of the axial thermal gradient analysis produced a temperature adjacent to the concrete of 145°F thereby demonstrating compliance with an imposed 150°F limit.

The above results are well within Code allowable stresses taken as 3 Sm due to the inclusion of discontinuity or secondary stress effects. Therefore, the flued heads have been demonstrated to comply with the design information supplied.

DATE OF ANALYSIS = 09/07/72

HEAD TYPE 7 PEN26(MODIFIED) CHECKOUT

DRAWING DETAIL NUMBER 112972 26

MATERIAL SELECTIONS BY TYPE NUMBER

HEAD	1
PROCESS PIPE	7
GUARD PIPE	0
NOZZLE PIPE	0
SUPPORT RING	-0

TABLE 1 - ALLOWABLE STRESS-INTENSITY(SM) (*10**3) (PSI)

SOURCE - USAS B31.7 (1969) NUCLEAR POWER PIPING TABLE A.1

MATERIAL NR. 1 - STAINLESS STEEL FORGINGS

TYPE - A182 F304

TEMP (F)	STRESS(SM)
70.0	20.0000
200.0	20.0000
300.0	19.8000
400.0	17.6000
500.0	16.4000
600.0	15.6000
650.0	15.3000
700.0	15.1000
750.0	14.9000
800.0	14.8000

TABLE 2 - MODULUS OF ELASTICITY(E) (*10**6) (PSI)

SOURCE - USAS B31.7 (1969) NUCLEAR POWER PIPING TABLE A.6

MATERIAL NR. 1 - STAINLESS STEEL FORGINGS

TYPE - A182 F304

TEMP (F)	** E **
70.0	28.3000
200.0	27.7000
300.0	27.1000
400.0	26.6000
500.0	26.1000
600.0	25.4000
700.0	24.8000
800.0	24.1000

TABLE 3 - COEFFICIENT OF THERMAL EXPANSION(ALPHA) (*10**-6)

SOURCE - USAS B31.7 (1969) NUCLEAR POWER PIPING TABLE A.5

MATERIAL NR. 1 - STAINLESS STEEL FORGINGS

TYPE - A182 F304

TEMP (F)	*ALPHA*
70.0	9.1100
100.0	9.1[illegible]00
150.0	9.2[illegible]00
200.0	9.3400
250.0	9.4100
300.0	9.4700
350.0	9.5300
400.0	9.5900
450.0	9.6500
500.0	9.7000
550.0	9.7600
600.0	9.8200
650.0	9.8700
700.0	9.9300
750.0	9.9900
800.0	10.0500

TABLE 4 - THERMAL CONDUCTIVITY(K)
(BTU/HR.-FT.-F)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
TABLE A.4

MATERIAL NR. 1 - STAINLESS STEEL FORGINGS

TYPE - A182 F304

TEMP (F)	** K **
70.0	8.3400
100.0	8.4000
150.0	8.6700
200.0	8.9000
250.0	9.1200
300.0	9.3500
350.0	9.5600
400.0	9.[illegible]000
450.0	10.0000
500.0	10.2300
550.0	10.4500
600.0	10.7000
650.0	10.9000
700.0	11.1000
750.0	11.3500
800.0	11.5500

TABLE 5 - THERMAL DIFFUSIVITY (K/CP)
((FT**2)/HR.)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
TABLE A.4

MATERIAL NR. 1 - STAINLESS STEEL FORGINGS

TYPE - A182 F304

TEMP (F)	* K/CP *
70.0	.1498
100.0	.14[illegible]5
150.0	.1525
200.0	.154[illegible]
250.0	.1565
300.0	.1[illegible]9
350.0	.1601
400.0	.1630
450.0	.1639
500.0	.165[illegible]
550.0	.16[illegible]4
600.0	.1707
650.0	.1721
700.0	.173[illegible]
750.0	.1757
800.0	.1778

TABLE 6 - ALTERNATING STRESS AMPLITUDES(SOURCE - USAS B31.7 (1969)

SA) (PSI) NUCLEAR POWER PIPING
FIG 1-705 3.3(A),(B)

MATERIAL NR. 1 - STAINLESS STEEL FORGINGS

TYPE - A182 F304

STRESS(SA)	N(CYCLES)
26000.	1000000.0
37500.	100000.0
59500.	10000.0
109000.	1000.0
240000.	100.0
650000.	10.0

TABLE 1 - ALLOWABLE STRESS-INTENSITY(SM) SOURCE - USAS B31.7 (1969)
(*10**3) (PSI) NUCLEAR POWER PIPING
TABLE A.1

MATERIAL NR. 7 - STAINLESS STEEL SEAMLESS PIPE

TYPE - A312 TP304

TEMP (F)	STRESS(SM)
70.0	20.0000
200.0	20.0000
300.0	19.0000
400.0	17.6000
500.0	16.4000
600.0	15.6000
650.0	15.3000
700.0	15.1000
750.0	14.9000
800.0	14.8000

TABLE 2 - MODULUS OF ELASTICITY(E) (*10**6) (PSI)

SOURCE - USAS B31.7 (1969) NUCLEAR POWER PIPING TABLE A.6

MATERIAL NR. 7 - STAINLESS STEEL SEAMLESS PIPE

TYPE - A312 TP304

TEMP (F)	** E **
70.0	28.3000
200.0	27.7000
300.0	27.1000
400.0	26.6000
500.0	26.1000
600.0	25.4000
700.0	24.8000
800.0	24.1000

TABLE 3 - COEFFICIENT OF THERMAL EXPANSION(ALPHA) (*10**-6)

SOURCE - USAS B31.7 (1969) NUCLEAR POWER PIPING TABLE A.5

MATERIAL NR. 7 - STAINLESS STEEL SEAMLESS PIPE

TYPE - A312 TP304

TEMP (F)	*ALPHA*
70.0	9.1100
100.0	9.1400
150.0	9.2500
200.0	9.3400
250.0	9.4100
300.0	9.4700
350.0	9.5300
400.0	9.5900
450.0	9.6500
500.0	9.7000
550.0	9.7600
600.0	9.8200
650.0	9.8700
700.0	9.9300
750.0	9.9900
800.0	10.0400

TABLE 4 - THERMAL CONDUCTIVITY(K) (BTU/HR.-FT.-F)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
TABLE A.4

MATERIAL NR. 7 - STAINLESS STEEL SEAMLESS PIPE

TYPE - A312 TP304

TEMP (F)	** K **
70.0	8.3500
100.0	8.4000
150.0	8.6700
200.0	8.9000
250.0	9.1200
300.0	9.3500
350.0	9.5600
400.0	9.8600
450.0	10.0000
500.0	10.2300
550.0	10.4500
600.0	10.7000
650.0	10.9000
700.0	11.1000
750.0	11.3500
800.0	11.5500

TABLE 5 - THERMAL DIFFUSIVITY (K/CP) ((FT**2)/HR.)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
TABLE A.4

MATERIAL NR. 7 - STAINLESS STEEL SEAMLESS PIPE

TYPE - A312 TP304

TEMP (F)	* K/CP *
70.0	.1498
100.0	.1495
150.0	.1525
200.0	.1548
250.0	.1568
300.0	.1589
350.0	.1601
400.0	.1639
450.0	.1639
500.0	.1659
550.0	.1684
600.0	.1707
650.0	.1721
700.0	.1736
750.0	.1757
800.0	.1778

TABLE 6 - ALTERNATING STRESS AMPLITUDES(SA) (PSI)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
FIG 1-705 3.3(A).(B)

MATERIAL NR. 7 - STAINLESS STEEL SEAMLESS PIPE

TYPE - A312 TP304

STRESS(SA)	N(CYCLES)
26000.	1000000.0
37500.	100000.0
59500.	10000.0
100000.	1000.0
240000.	100.0
650000.	10.0

TABLE 1 - ALLOWABLE STRESS-INTENSITY(SM) (*10**3) (PSI)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
TABLE A.1

MATERIAL NR. 9 - CARBON STEEL SEAMLESS PIPE

TYPE - A106 GR.B

TEMP (F)	STRESS(SM)
70.0	20.0000
400.0	20.0000
500.0	19.9000
600.0	17.3000
650.0	17.0000
700.0	16.8000

TABLE 2 - MODULUS OF ELASTICITY(E) (*10**6) (PSI)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
TABLE A.6

MATERIAL NR. 9 - CARBON STEEL SEAMLESS PIPE

TYPE - A106 GR.B

TEMP (F)	** E **
70.0	27.9000
200.0	27.7000
300.0	27.4000
400.0	27.0000
500.0	26.4000
600.0	25.7000
700.0	24.8000

TABLE 3 - COEFFICIENT OF THERMAL EXPANSION(ALPHA) (*10**-6)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
TABLE A.5

MATERIAL NR. 9 - CARBON STEEL SEAMLESS PIPE

TYPE - A106 GR.B

TEMP (F)	*ALPHA*
70.0	6.0700
100.0	6.1300
150.0	6.2[illegible]00
200.0	6.3800
250.0	6.4900
300.0	6.6000
350.0	6.7100
400.0	6.8200
450.0	6.9200
500.0	7.0200
550.0	7.1200
600.0	7.2300
650.0	7.3300
700.0	7.4400
750.0	7.5[illegible]00
800.0	7.6[illegible]00

TABLE 4 - THERMAL CONDUCTIVITY(K)
(BTU/HR.-FT.-F)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
TABLE A.4

MATERIAL NR. 9 - CARBON STEEL SEAMLESS PIPE

TYPE - A106 GR.B

TEMP (F)	** K **
70.0	36.0000
100.0	36.6000
150.0	35.7000
200.0	34.8500
250.0	34.0000
300.0	33.2000
350.0	32.3000
400.0	31.6000
450.0	30.6000
500.0	29.8000
550.0	29.0000
600.0	28.3000
650.0	27.5000
700.0	26.8000
750.0	26.0000
800.0	25.4000

TABLE 5 - THERMAL DIFFUSIVITY (K/CP)
((FT**2)/HR.)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
TABLE A.4

MATERIAL NR. 9 - CARBON STEEL SEAMLESS PIPE

TYPE - A106 GR.B

TEMP (F)	* K/CP *
70.0	.6505
100.0	.6552
150.0	.6345
200.0	.6094
250.0	.5849
300.0	.5622
350.0	.5347
400.0	.5191
450.0	.4953
500.0	.4753
550.0	.4569
600.0	.4387
650.0	.4173
700.0	.3986
750.0	.3791
800.0	.3606

TABLE 6 - ALTERNATING STRESS AMPLITUDES(SA) (PSI)

SOURCE - USAS B31.7 (1969)
NUCLEAR POWER PIPING
FIG I-705 3.3(A).(B)

MATERIAL NR. 9 - CARBON STEEL SEAMLESS PIPE

TYPE - A106 GR.B

STRESS(SA)	N(CYCLES)
12500.	1000000.0
20000.	100000.0
3[illegible]000.	10000.0
83000.	1000.0
20[illegible]000.	100.0
5[illegible]0000.	10.0

***** T H E R M A L A N A L Y S I S *****

1 STEADY STATE CASES TO BE ANALYZED

0 TRANSIENT CASES TO BE ANALYZED

INSULATION JACKET DATA

THICKNESS OF INSULATION (INCHES) = 1.5000
EMMISSIVITY OF OUTER INSULATION SURFACE = .30000000E+00
NUMBER OF LONGITUDINAL CAVITY INSULATION SEGMENTS = 1

INSULATION CONDUCTIVITY TABLE

TEMPERATURE (F)	CONDUCTIVITY (BTU-IN/FT2-F)
40.000	.30000000E+00
300.000	.42000000E+00
400.000	.51000000E+00
700.000	.64000000E+00

THERMAL BOUNDARY CONDITIONS

INITIAL TEMPERATURE OF MODEL (F) = 200.000
AMBIENT AIR TEMPERATURE (F) = 95.000

PROCESS LINE FLUID IS WATER (TYPE 1)

PRESSURE OF WATER (PSIG) = 2200.000
VELOCITY OF WATER (FT/SEC.) = 6.500

STEADY STATE ANALYSIS

PROCESS PIPE FLUID TEMPERATURE (F)	=	450.000
GUARD PIPE TEMPERATURE (F)	=	-0.000
GUARD PIPE TEMPERATURE LENGTH (IN)	=	-0.000
WALL SINK TEMPERATURE (F)	=	[illegible].000
WALL LENGTH OF NOZZLE (IN)	=	1[illegible].000
DIAMETRICAL DISTANCE OF WALL SINK (IN)	=	48.000
MOUNTING RING EXTREMETY TEMPERATURE (F)	=	-0.000

RADIATION RESISTOR DATA - FUNCTION 2

STEADY STATE CASE 1

STEADY STATE GRADIENT

EXTERNAL NODE TEMPERATURES (F)

PROCESS LINE	= 450.0000
AMBIENT AIR	= 95.0000
CAVITY INSULATION NODE 1 (Z= 35.53)	= 234.5997
EXTERIOR INSULATION	= 187.7676
CONCRETE WALL	= 95.0000

INTERNAL NODE TEMPERATURES (F)

1	449.7158	2	449.8590	3	449.4894	4	449.7296	5	442.6436
6	443.7476	7	446.6279	8	440.2177	9	442.7420	10	446.3671
11	435.1357	12	437.8408	13	440.9499	14	445.6179	15	428.5140
16	431.7874	17	436.7615	18	443.5857	19	449.8864	20	449.7764
21	449.8864	22	449.7764	23	449.8864	24	449.7764	25	449.8864
26	449.7764	27	449.8864	28	449.7764	29	449.8864	30	449.7764
31	449.8864	32	449.7764	33	449.8864	34	449.7764	35	449.8864
36	449.7764	37	449.8864	38	449.7764	39	449.8864	40	449.7764
41	449.8864	42	449.7764	43	449.8864	44	449.7764	45	449.8842
46	449.7700	47	449.8582	48	449.7212	49	448.2180	50	446.5605
51	445.9735	52	447.9435	53	445.9991	54	444.6794	55	447.2239
56	444.4119	57	442.6369	58	445.4431	59	440.8728	60	438.7343
61	436.8318	62	442.7437	63	435.1533	64	430.1912	65	427.3913
66	437.3771	67	423.8011	68	414.7196	69	439.6312	70	441.1131
71	418.6689	72	436.0217	73	433.1883	74	430.0380	75	426.8300
76	425.2218	77	424.2485	78	429.1326	79	432.3747	80	439.9631
81	425.4869	82	419.7325	83	413.9695	84	407.7196	85	400.7220
86	397.2652	87	399.8799	88	416.2914	89	412.8175	90	422.7136
91	414.9515	92	406.4888	93	398.2923	94	390.0738	95	378.5222
96	373.0843	97	377.6216	98	388.3318	99	397.8458	100	417.3790
101	407.9405	102	396.6618	103	385.1663	104	374.7615	105	355.5036
106	349.0461	107	354.7330	108	372.3969	109	387.9165	110	325.3816
111	325.1197	112	325.9641	113	314.1711	114	315.9421	115	306.7523
116	288.4266	117	289.5664	118	287.5723	119	280.0371	120	276.7722
121	276.5716	122	277.3495	123	276.0209	124	273.8971	125	272.9282
126	269.1253	127	268.8523	128	267.9495	129	269.4467	130	269.6378
131	263.4618	132	263.5657	133	272.8193	134	276.2557	135	269.2482
136	268.4961	137	192.3369	138	191.9654	139	192.6621	140	192.6625
141	193.9140	142	193.8359	143	196.3423	144	196.2555	145	199.4459
146	199.3576	147	203.9139	148	203.8251	149	209.6144	150	209.5271
151	217.0950	152	217.0525	153	225.0600	154	225.6504	155	232.6605
156	232.4561	157	239.4414	158	239.4156	159	245.9098	160	245.9936
161	260.8975	162	264.3247	163	262.0276	164	259.6262	165	268.5729
166	264.5151	167	264.4229	168	236.9188	169	236.0248	170	211.6464
171	211.7262	172	141.5966	173	141.4657	174	141.9991	175	142.0459
176	142.6154	177	142.4869	178	143.1522	179	143.1917	180	143.9619
181	143.8399	182	144.6774	183	144.7074	184	145.6615	185	145.4821
186	146.2846	187	146.3056	188	146.8825	189	146.7618	190	146.8825

APPENDIX 3G5

REPORT NO. 71983-6

T-700-

TUBE TURNS
LOUISVILLE 1, KENTUCKY

ENGINEERING DEPARTMENT

REPORT NO. 71983-6

FLUED HEAD

STRESS REPORT

FOR

FLORIDA POWER AND LIGHT COMPANY

ST. LUCIE PLANT

(FORMERLY HUTCHINSON ISLAND)

CONTAINMENT PIPING PENETRATION ASSEMBLIES

P.O. NO. NY-422264

Included Are:

Penetration No. 1, Main Steam (SG-1A), I-34MS-28
Penetration No. 3, Feedwater (SG-1A), I-20BF-14

Prepared by: WS Haberman Date: 7 AUG 73
W.S. Haberman

Approved by: M.V. Malkmus Date: 8/7/73
M.V. Malkmus

Although this stress report is believed to be accurate, nothing contained herein shall be construed as establishing a warranty, express or implied.

T-700-

TUBE TURNS
LOUISVILLE 1, KENTUCKY

ENGINEERING DEPARTMENT

REPORT NO.

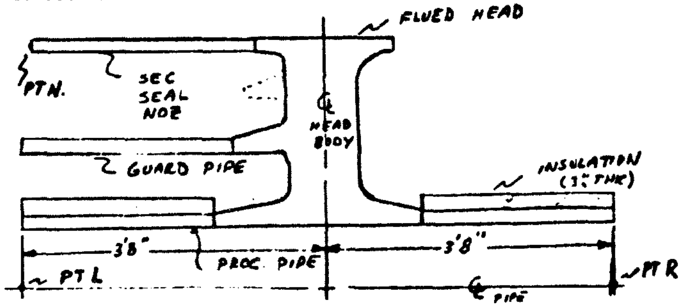
Report Summary:

Introduction

The following, when combined with the balance of items listed in the index, constitutes the stress report on the two (2) flued head forgings configurations utilized in the Type I penetration piping assemblies for St. Lucie No. 1 (formerly Hutchinson Island). All calculations for stress distributions, herein, were made using finite element computer techniques. Descriptive data relative to the program and its verification is found in Section I-3 and I-4 of Report No. 71983-5, previously submitted.

Basic Approach

For analysis purposes the flued head and attached piping model was assumed as shown below:



The external load combinations considered are shown below. These were specified by Ebasco in a memo dated December 22, 1972 and are assumed to act at the flued head centerline and can be produced by either the right or left hand process pipe inthe worst combination stresswise:

Main Steam

Load Combinations	Axial (KIPS)	Trans. (KIPS)	Bending Moment (FT-KIPS)	Torsion (FT-KIPS)	Case No.
Thermal & Seismic	120	-	100	450	1
(O.B.E.)	40	60	1300	450	2
	40	60	100	800	3
Thermal & Seismic	200	-	-	-	4
(D.B.E.	40	120	100	450	5

MAIN STEAM

Load Combinations	AXIAL (KIPS)	TRANS. (KIPS)	BENDING MOMENT (FT.KIPS)	TORSION (FT.KIPS)	CASE NO.
Thermal & Seismic	40	120	2500	450	6
(D.B.E.)	-	120	100	1150	7

TUBE TURNS
LOUISVILLE 1, KENTUCKY

ENGINEERING DEPARTMENT

REPORT NO. 71983-6

FEEDWATER

Load Combinations	Axial (KIPS)	Trans (KIPS)	Bending Moment (FT.KIPS)	Torsion (FT.KIPS)	Case NO.
Thermal & Seismic (O.B.E.)	44	60	220	10	1
	19	79	320	10	2
	19	79	220	110	3
Thermal & Seismic (D.B.E.)	69	60	220	10	4
	19	98	420	10	5
	19	98	220	210	6

For the actual stress runs, the above loads were converted to statically equivalent loads applied to the ends of the analytical model shown previously. The actual loading data used is shown below:

MAIN STEAM (P1, P2)

Load	1	2	3	4	5	6	7
F_R	-120000	-40000	-40000	-200000	-40000	-40000	-
F_L	-120000	-40000	-40000	-200000	-40000	-40000	-
V_L	-	60000	60000	-	120000	120000	-
M_R	-1200000	-15600000	-1200000	-	-1200000	-30000000	1200[illegible]
M_L	-1200000	-12960000	1440000	-	4080000	-24720000	1200[illegible]
T_R	-5400000	-5400000	-9600000	-	-5400000	-5400000	-1380[illegible]
T_L	-5400000	-5400000	-9600000	-	-5400000	-5400000	-1380[illegible]
V_N	-27,273	-384500	-57280	-	-87273	-741800	-2727[illegible]
PRESS	885PSIG	885PSIG	885PSIG	885PSIG	885PSIG	885PSIG	885PS[illegible]
S.S. THERMAL	520°F	520°F	520°F	520°F	520°F	520°F	520°F

TUBE TURNS
LOUISVILLE 1, KENTUCKY

ENGINEERING DEPARTMENT

REPORT NO. 71983-6

FEEDWATER (P3, P4)

LOAD	1	2	3	4	5	6
F_R	-44000	-19000	-19000	-69000	-19000	-19000
F_L	-44000	-19000	-19000	-69000	-19000	-19000
V_L	60,000	79000	79000	60000	98000	98000
M_R	-2640000	-3840000	-2640000	2640000	-5040000	-2640000
M_L	0	-364000	836000	0	-728000	1672000
T_R	-120000	-120000	-1320000	-120000	-120000	-2520000
T_L	-120000	-120000	-1320000	-120000	-120000	-2520000
V_N	-90000	-126770	-99500	-90000	-163550	-109000
PRESS	1050PSIG	1050PSIG	1050PSIG	1050PSIG	1050PSIG	1050PSIG
S.S. Thermal	440°F	440°F	440°F	440°F	440°F	440°F

The nomenclature and sign convention for the above is as follows:

P_R = Axial load rt. end of process pipe (lb.)

P_L = Axial load left end of process pipe (lb.)

V_L = Transverse load left end of process pipe (lb.)

M_R = Bending moment right end of process pipe (in.lb.)

M_L = Bending moment left end of process pipe (in.lb.)

T_R = Torsional moment right end of process pipe (in.lb.)

T_L = Torsional moment left end of process pipe (in.lb.)

V_N = Transverse load outer nozzle (lb.)

To arrive at the final product a set of unit run cases was produced followed by a set of combination routines using appropriate multipliers.

It should be noted pipe rupture conditions and the balance of major penetration components have been evaluated in Reports 71983-1 and 71983-4.

Summary of Results

The results of the two (2)analyses are summarized below:

Main Steam (P1, P2)

Combination	Max.Stress Intensity(PSI)	Stress Location	Allowable Stress(PSI
1	26203	E1#90-Head Body I.D. Left Side	61141
2	36732	E1#82 O.D. Left Process Pipe Hub	61080
3	25806	E1#90 Head Body I.D. Left Side	61141
4	26110	E1#90 Head Body I.D. Left Side	61141

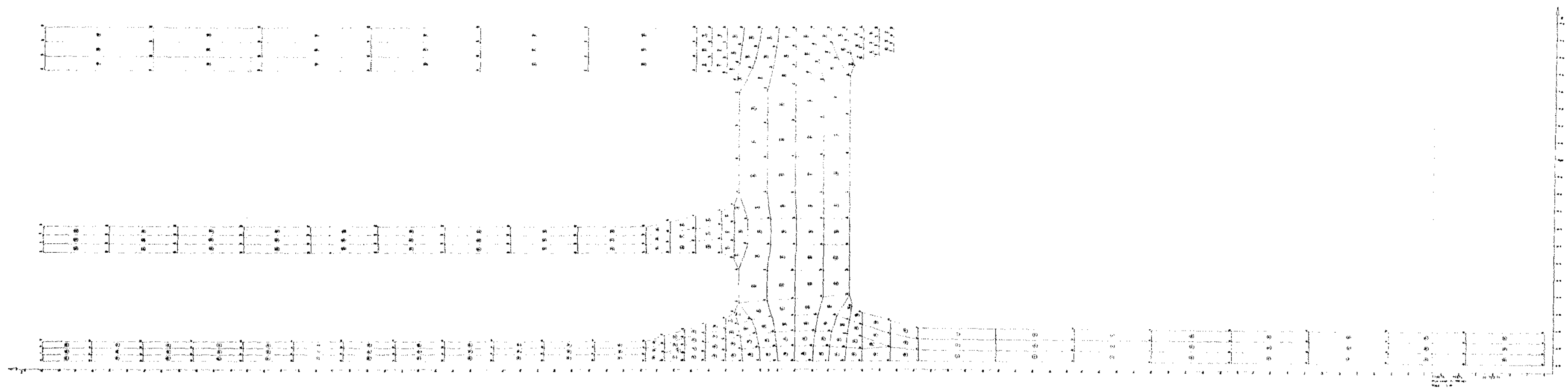
3G-64

MAIN STEAM (P1,P2)

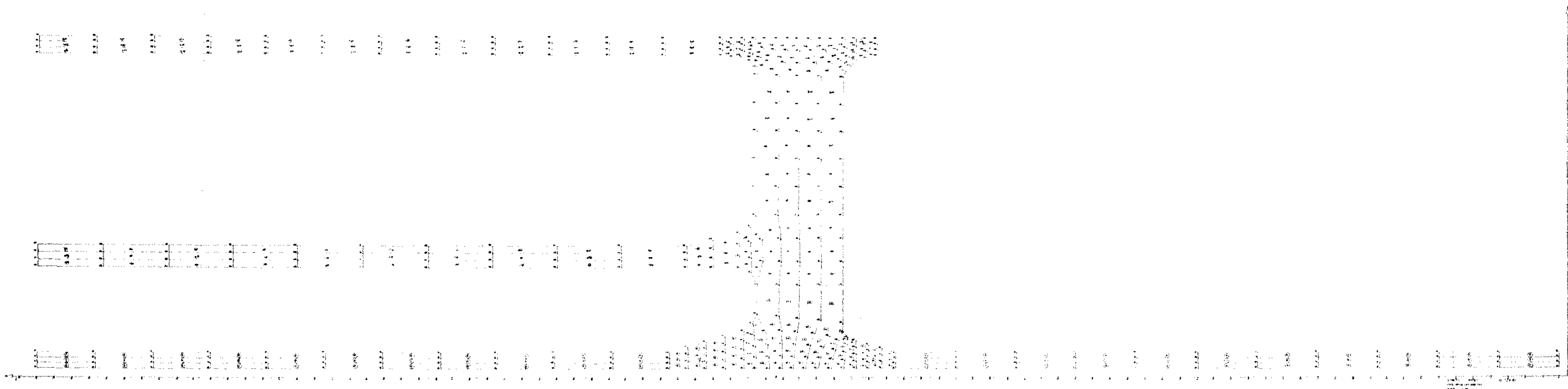
Combination	Max. Stress Intensity(PSI)	Stress Location	Allowable Stress(PSI)
5	25950	E1#90 Head Body I.D. Left Side	61141
6	57626	E1#114 O.D. Left Process Pipe Hub	61442
7	25338	E1#90 Head Body ID Left Side	61141

FEEDWATER (P3,P4)

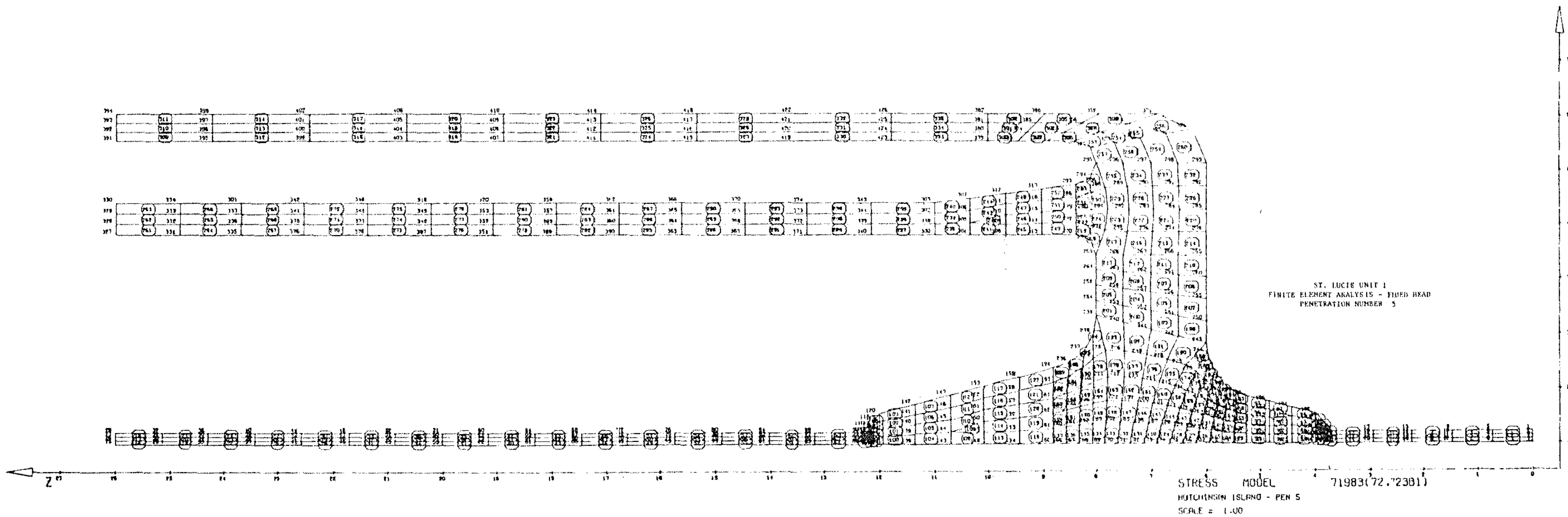
Combination	Max. Stress Intensity(PSI)	Stress Location	Allowable Stress(PSI)
1	32205	E1#100 Head Body I.D. Left Side	60520
2	34783	E1#101 Head Body I.D. Left Side	60473
3	31736	E1#100 Head Body I.D. Left Side	60520
4	32984	E1#101 Head Body I.D. Left Side	60473
5	38188	E1#101 Head Body I.D. Left Side	60473
6	31944	E1#100 Head Body I.D. Left Side	60520



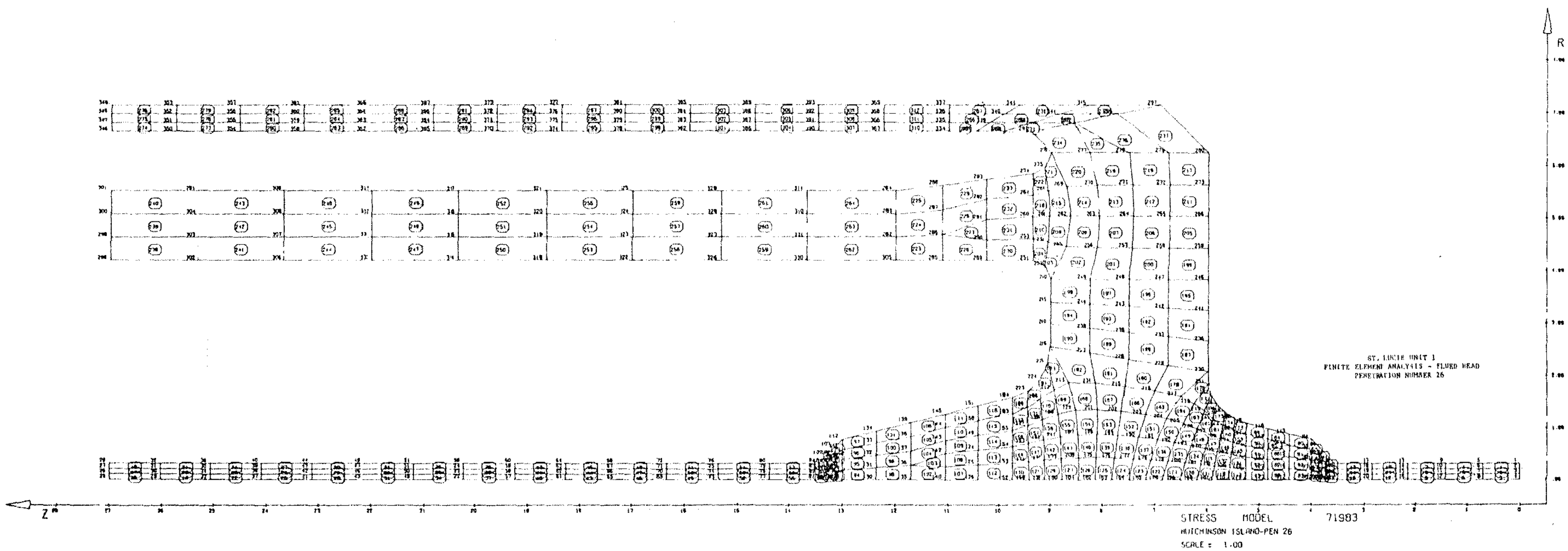
ST. LUCIE UNIT 1
FINITE ELEMENT ANALYSIS - FLUED HEAD
PENETRATION NUMBER 1 & 2



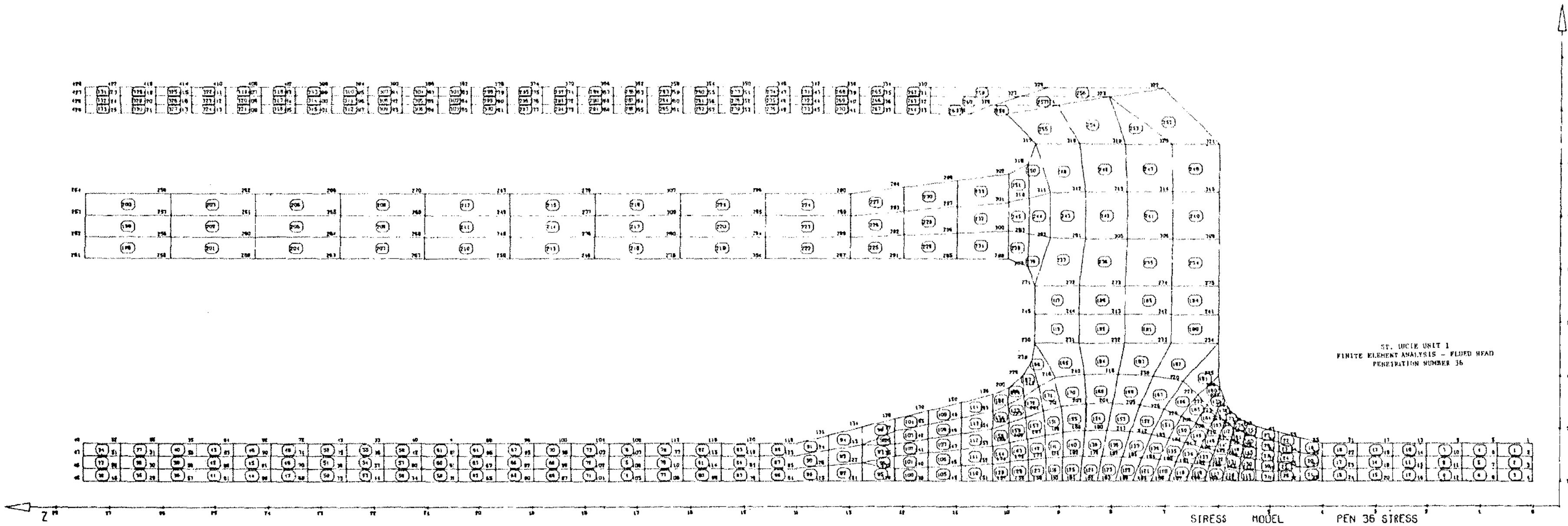
ST. LUCIE UNIT 1
FINITE ELEMENT ANALYSIS - FLUED HEAD
PENETRATION NUMBER 3 & 4



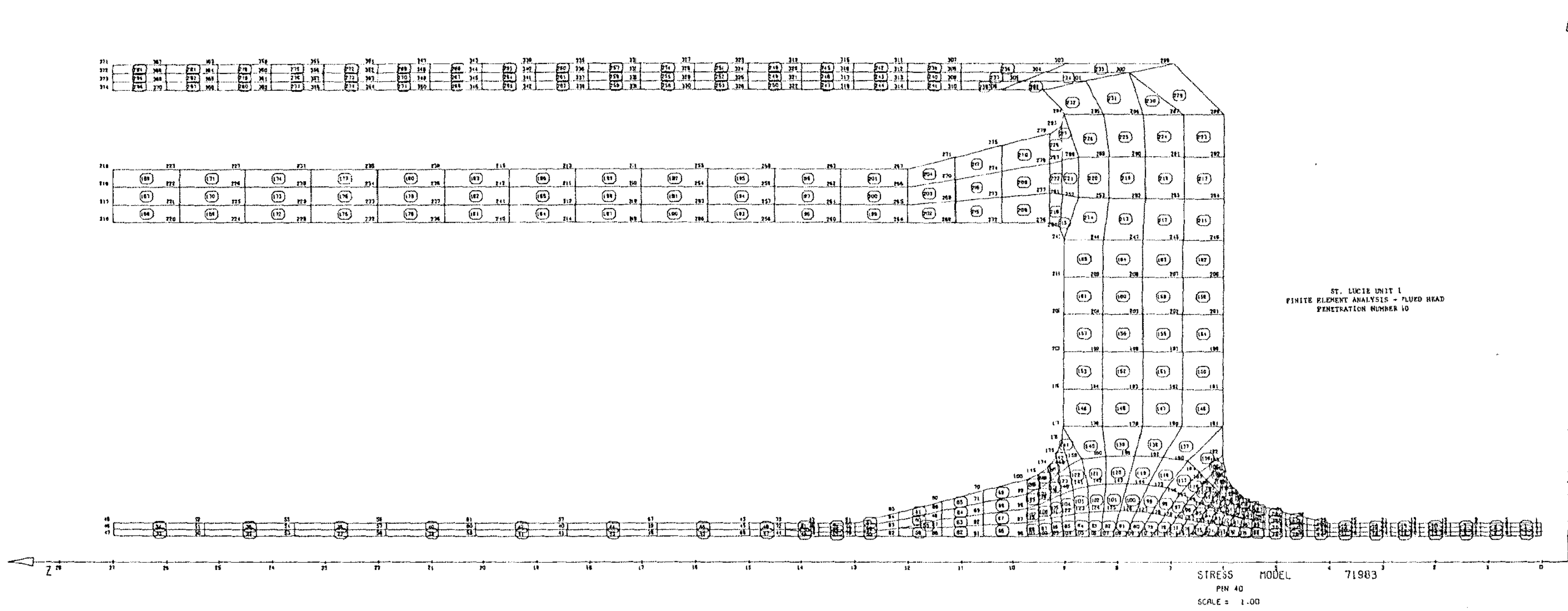
R
Z
ST. LUCIE UNIT 1
FINITE ELEMENT ANALYSIS - FIXED HEAD
PENETRATION NUMBER 5
STRESS MODEL 71983(72,72301)
HUTCHINSON ISLAND - PEN 5
SCALE = 1.00



ST. LUCIE UNIT 1
FINITE ELEMENT ANALYSIS - FLUED HEAD
PENETRATION NUMBER 26
STRESS MODEL 71983
HUTCHINSON ISLAND-PEN 26
SCALE : 1.00



STRESS MODEL PEN 36 STRESS
HUTCHINSON ISLAND - PEN36
SCALE = 1.00


ST. LUCIE UNIT 1
FINITE ELEMENT ANALYSIS - FLUED HEAD
PENETRATION NUMBER 10
R
Z
STRESS MODEL 71983
PIN 40
SCALE = 1.00

APPENDIX 3H

ANALYSIS OF REACTOR SUPPORT STRUCTURE

ST. LUCIE PLANT

APPENDIX 3H

TABLE OF CONTENTS

A. INTRODUCTION

This report is a summation of analytical work which was performed to review the structural adequacy of the reactor support system for St. Lucie Unit 1. The study was undertaken when increased reactor support loads and reactor pressure were determined from the use of more sophisticated multi-nodal pipe break analysis. Refined, more sophisticated thermal analysis techniques were used to study other design conditions including "loss of cooling fans."

The analysis performed in this study is based on an analytical representation of the support structure and the effects of the various physical parameters, such as temperature, applied loads and pressures, on the structure. The analytical work used established computer analysis methods. All parameters were based on an evaluation of relevant conditions within the containment.

The results of analyses determine that the reactor support system, consisting of the steel beam-column assembly and concrete primary shield wall will withstand the combination of loads postulated for the plant design and that the support deformations are within allowable limits. Specifically, it demonstrates that the reactor cavity and reactor support system can accommodate the highly unlikely pipe ruptures postulated concurrent with maximum seismic loadings.

With regard to the recently defined loads (NRC letter to FP&L of November 28, 1975) on the reactor supporting system resulting from asymmetric loads on the core barrel subsequent to a postulated guillotine break at the cold leg nozzle the applicant will perform an analysis of the reactor vessel support system. This load will be analyzed in conjunction with the previously analyzed pipe reaction and external pressure vessel asymmetric loads to:

(1) determine the loads in the reactor vessel support system

(2) evaluate the full restraint capability of the support system

(3) compute the safety margins of the support system.

This analysis will be completed as expeditiously as practicable. Our current schedule indicates a completion date of August 1977.*

---

* A report entitled "Reactor Support System - Evaluation of Margins" (See Reference 1) was transmitted to the NRC assessing the margin in design of the vessel supports when the internal asymmetric loads are added to all previous loads. The report concluded that the supports would adequately withstand all the loadings. See Section 3.6.3.1.

B. GENERAL DESCRIPTION

B.1. Analysis

The original analyses and design of the reactor support structures were based on criteria established for loading conditions, thermal limitations, and allowable stress levels. Conservative traditional stress analysis methods were utilized for the designs which were completed in early 1972. The outline of the resulting concrete and steel structures are shown in Figures 3H-1 and 2. As an integral part of the design, reactor support cooling system was designed to provide direct cooling to the support girders. A reactor cavity cooling system provides cooling for the reactor cavity area in general. Both ventilation systems are described in Section 9.4.

B.2 Reactor Support Structure

The reactor is supported at three points on a steel girder-column assembly within the reactor cavity. The ends of the girder are embedded in the 7 ft - 3 in. thick concrete primary shield wall. The column is bolted to the underside of the girder and to the reactor cavity floor. The support arrangement is shown in Figures 3H-1 and 2. The other points of support for the reactor system are at the steam generators.

Load transfer between the reactor system and the reactor support occur between the support shoe which is welded to the reactor nozzle and steel bearing plates designed into the top of the steel support girder. The support shoe is free to slide in a direction longitudinally along the axis of the nozzle and only a frictional load is transmitted to the support structure in this direction. In the transverse direction, steel plates at the top of the girder take load from the shoe in direct bearing. Downwardly acting loads are transmitted directly into the supporting girder.

The reactor support loads associated with the various critical loading combinations are shown in Figures 3H-3,4 and 5. The loading combinations are described in Section C.

The arrangement of the reactor building internal concrete is shown in Figures 3H-6 and 7. The internal structure consists of a 33 ft thick section of mass concrete which rests atop the 10 ft thick reactor building mat. The reactor cavity extends about 21 ft into the mass concrete section. Above this, a 7 ft - 3 in. thick concrete primary shield wall surrounds the reactor. This wall continues up to the operating deck and forms a part of the refueling canal wall. The reactor support beams are embedded into the primary shield wall about 5 ft above the top of the mass concrete.

B.3 Ventilation Cooling Systems

B.3.a Reactor Support Cooling System

The reactor support cooling system ventilates the reactor support girders so that the interface temperature between girder and concrete does not exceed 120°F during normal operation. The reactor support leg face temperature is 300°F. The reactor support cooling system ventilates the reactor support girders and removes heat which is transmitted from the reactor support shoe.

Ventilation pipes disperse air at selected points over the girder to maintain an acceptable temperature gradient along the girder length and restrict the vertical thermal growth of the supporting steel work for reactor vessel to 3/16 of an inch relative to steam generator sliding base support. The ventilation scheme is shown in Figure 3H-8.

B.3.b Reactor Cavity Cooling System

The reactor cavity cooling system ventilates the annular space between the reactor vessel and primary shield in order to limit concrete temperature to 150°F during normal operation.

The duct system works as follows: a duct connection from the containment ring header conveys cool air at 105°F to the inlets of two 100 percent capacity redundant fans. The fans discharge this air through a connecting duct to a ring header at the bottom of the cavity. The air supplied from outlets on this header sweeps upward through the annular reactor cavity to the containment operating floor. Refer to Figure 3H-8.

C. LOADS AND LOADING COMBINATIONS

The loading combinations listed in Section 3.8.3.3.2, "Loading Combinations," were used in the analysis of the reactor support system. These load combinations, established for the design in 1969, are essentially identical to those outlined in Enclosure 2, "Structural Design Criteria for Evaluating the Effects of High Energy Pipe Breaks in Category I Structures Outside the Containment," forwarded to the Applicant via AEC letter dated August 29, 1973. The critical loading combination (LOCA + DBE) is identical.

Each loading combination is used to determine a final load for the design of a particular structural element. The actual design is based on the highest final (critical) load determined via these combinations. After inspection of all load combinations, analyses were performed for the loading combinations listed hereinafter.

C.1 Concrete Structures

C.1.a Loading Combinations

The reactor support system is designed to withstand the effects of normal operation, design basis earthquake, and maximum credible accidents (including LOCA).

The following loading combinations have been used in the analysis of the concrete primary shield wall:

a) Normal operating, steady state (120° F)

$$U = 1.5\,(D+L' + A + T)$$

b) Normal operating, steady state with OBE

$$U = 1.5\,(D+L' + A + T_1) + 1.5\,(OBE)$$

c) Normal operating, start-up transient (70° F to 120° F)

$$U = (D+L' + A + T_2)$$

d) Loss of coolant accident (LOCA) during the steady state condition (blowdown transient) with DBE.

$$U = T_1 + 1.15\,(D+L' + A + P + Q + DBE)$$

e) Post-LOCA (post blowdown)

$$U = D+L' + A + T_3$$

f) Loss of fans during the steady state condition (120° F)

$$U = D+L' + A + T_4$$

g) Loss of fans during initial start-up (70°F to 120°F)

$U = D + L' + A + T_5$

where

U	=	The ultimate strength of the concrete structural components required to resist the factored loads listed in (a) to (g).
D	=	Dead load consists of the dead weight of the reactor building internal structure, the weight of structure steel and miscellaneous building items within the containment vessel.
A	=	The pipe or equipment anchor loads are the loads exerted upon the various structural elements in the reactor building internal structure by the pipe or equipment restraints for normal thermal expansion of the various piping systems.
T_1	=	Moments and forces caused by the temperature distribution during normal operating, steady state condition.
T_2	=	Moments and forces caused by the temperature effects during initial transient for start-up to operating power conditions.
T_3	=	Moments and forces caused by the temperature effects during a post-LOCA condition.
T_4	=	Moments and forces caused by the temperature effects subsequent to a loss of fans is assumed to occur during the steady state condition.
T_5	=	Moments and forces caused by the temperature effects subsequent to a loss of fans during initial start-up.
OBE	=	Moments and forces caused by an operating basis earthquake.
DBE	=	Moments and forces caused by a design basis earthquake.
Q	=	These are the loads exerted upon the reactor building internal structure by a pipe or a piece of equipment as a result of a postulated LOCA or steam line break accident.
P	=	Pressure loads within the primary shield area, including jet impingement, resulting from a loss of coolant accident. The accident is assumed to occur during the steady state temperature condition.

The peak pressures resulting from the multi-nodal analysis are used. These peak pressures result in a maximum pressure differential across the cavity; an axisymmetrical pressure distribution is not assumed.

L' = The dead weight of the various pieces of equipment, including water, steam or the other enclosed fluids, supported by the reactor building internal structure.

Forces transmitted from the reactor coolant system to the reactor supports and resulting from L', A, Q and DBE are based on the critical loads shown in Table 3H-1. The accident loads listed in Table 3H-1 include a dynamic factor of 2, normal operating loads and maximum seismic loads. The direction of the seismic forces included in each accident load was chosen to result in the largest load at each component. Although "A" is a normal operating load, it has been used for loading combinations (a) through (g). Temperature distributions in the reactor support system caused by loads T_1, T_2, T_3, T_4, and T_5 are listed in Attachment 1.

Pressure loads, P, are based on Unit 1 FSAR, Revision 36, Figures 6.2-18a to 6.2-18r, 6.2-19a to 6.2-19r, 6.2-20a to 6.2-20r.

C.1.b <u>Acceptance Criteria</u>

The acceptance criteria given in Section 3.8.3, "Containment Concrete Internal Structures and Seismic Category I Structural Steel," is applicable to the analysis work just completed on the reactor support system with the following exception for concrete:

Section 1503 of ACI 318-63, wherein tensile strength of concrete in flexual members is neglected, has not been followed in this analysis. Instead, the tensile capacity of the concrete has been included in the analysis of the cavity wall as follows:

The primary shield wall is 7 feet - 3 inches thick. The reinforcing arrangement consists of horizontal hoop reinforcing and vertical reinforcing in both faces of the wall in addition to layers of hoop reinforcing and a layer of transverse reinforcing provided under the sleeves provided for the six primary pipe lines. Additional vertical reinforcing, including ties, are carried between the penetrations. The reinforcing arrangement is shown in Figures 3H-9, 10 and 11.

Interior areas of the wall which are not reinforced are analyzed by the comparing maximum principal stresses against an envelope of limiting stresses defining failure of plain concrete. This envelope is based on Mohr's Theory of Failure and is shown in Figure 3H-12. Overstressing is considered to have occurred when the Mohr's circle for an element, as defined by its principal stress output from the finite element analysis, exceeds the Mohr's failure envelope. When this occurs, the element is considered to have cracked. This cracked element is inputed in the finite element model, and the analysis is rerun. Because of the partially cracked finite element model, the rerun results in a new stress distribution which could result in new overstressed elements. The new overstressed elements are "cracked" and the model rerun.

Except as indicated above, all other criteria were based on ACI 318-63. The use of ACI 318-63, "Building Code Requirements for Reinforced Concrete Structures" is a standard code, accepted and used industry wide.

The use of an allowable tensile stress in plain concrete has been extensively studied and documented, and is recognized by various national engineering committees such as the American Concrete Institute. The criteria used in this analysis were derived principally from information provided in the publication "The Shear Strength of Concrete Members," prepared by the Joint ASCE-ACI Task Committee 426, from information contained in ACI Monograph No. 6, "Hardened Concrete: Physical and Mechanical Properties," and from ASCE Structural Division Paper 3036, "Strength of Reinforced Concrete Beams" by Signey Guralnide.

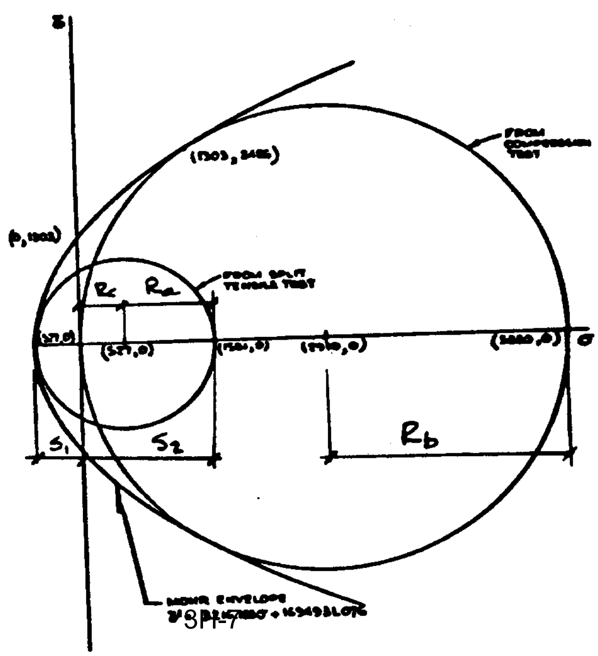
In addition, Splitting Tensile Strength (ASTM C-496), Modulus of Rupture (ASTM C-78) and Modulus of Elasticity (ASTM C-469) tests were made for the design concrete mix used for the primary shield wall. These test results confirm the allowable tensile strength used in the analysis and are in agreement with other test results documented in the above mentioned literature. The test results are shown in the attached Pittsburgh Testing Laboratory report dated 2-20-75.

The Mohr's failure envelope was developed as follows:

The Mohr's failure envelope is a curve as shown in Figure 3H-12 of the Mohr's stress design.

The curve is tangent to the Mohr's circle representing the stresses of the unconfined compression test (ASTM C-39) and the Mohr's circle representing the stresses of the tensile splitting test (ASTM C-496).

The maximum principal stresses are determined as follows:



SPLIT TEST

$$S_1 = -\frac{w}{\pi R} = \frac{-2w\ell}{2\pi R\ell} = \frac{-2}{\pi D\ell} = -527 \text{ psi}$$

$$S_2 = \frac{3wR^2}{\pi R^3} = \frac{3w}{\pi R} = \frac{6w}{2\pi R} = \frac{6P}{\pi D\ell} = 1581 \text{ psi}$$

Where $W = \frac{P}{\ell}$

$P = 59625$ lbs

$D = 6''$

$\ell = 12''$

COMPRESSION TEST

$f_c' = 5820$

$$R_a = \frac{S_2 + S_1}{2} = 1054, \quad R_a^2 = 1110916$$

$$R_b = \frac{fc'}{2} = 2910, \quad R_a^2 = 8468100$$

$$R_c = \frac{S_2 - |S_1|}{2} = 527$$

Rupture Circle A $\quad Y^2 = R_a^2 - (X - R_c)^2 \quad$ (1)

Compression Circle B $\quad Y^2 = R_b^2 - (X - R_b)^2 \quad$ (2)

The curve (failure envelope) has been fitted to these two circles graphically; the results are shown in Figure 3H-12.

C.2 Steel Structures

C.2.a Loading Combinations

The following loading combinations have been used in the analysis of the steel beam-column assembly:

a) Normal operation
$W = D + L' + T_1 + OBE + A$

b) Loss of coolant accident (blowdown transient)
$W = D + L' + T_1 + DBE + A + Q$

c) Post-LOCA (post blowdown)
$W = D+L' + A + T_3$

d) Loss of fans
$W = D+L' + A + T_4$

where

W = The total loading applied to the steel beam-column assembly.

$D+L'$ = Dead loads including the forces transmitted from the reactor coolant system to the reactor support system.

A = Concentrated forces due to temperature effects transmitted from the reactor coolant system to the reactor supports.

T_1 = The temperature distribution during normal operating condition.

T_3 = The temperature distribution during a post-LOCA.

T_4 = The temperature distribution subsequent to a breakdown of fans.

DBE = Forces caused by a design basis earthquake.

OBE = Forces caused by an operating basis earthquake.

Q = Concentrated forces acting on the reactor support system and caused by a postulated pipe break.

Note: Examination of the differential pressure outputs of the multi-nodal analysis shows that the loads imposed on the steel structures due to P are on the order of 50 kips. This magnitude is negligible in comparison with the other loads considered in the LOCA case (several thousand kips) and thus P was not included.

Forces transmitted from the reactor coolant system to the reactor supports and resulting from L', A, Q and DBE are based on the critical loads shown in Table 3H-1. The accident loads listed in Table 3H-1 include a dynamic factor of 2, normal operating loads and maximum seismic loads. The direction of the seismic forces included in each accident load was chosen to result in the largest load at each component. Although "A" is a normal operating load, it has been used for loading combinations (a) through (d). Temperature distributions in the reactor support system caused by loads T_1, T_3 and T_4 are listed in Attachment 1.

The above loading combinations have been evaluated for both the start-up transient and steady state conditions.

C.2.b Acceptance Criteria

Normal and maximum allowable stresses have been used as permitted by A.I.S.C. These stress allowables are shown in Table 3H-2.

D ANALYSIS

Analyses have been performed separately for the steel beam-column assembly taken as an independent frame and subjected to the various loading conditions described previously, and for the entire reactor support system comprised of the lower mass concrete, primary shield wall and steel beam-column assembly and similarly subjected to the various loading conditions described. The steel beam-column assembly analysis as described in Section D.2 is based on the conservative assumption that the beam is fully restrained from thermal expansion in all loading conditions considered. In other words, a thermal design load is utilized in the steel beam-column analysis for a fully restrained (no movement) beam based on temperature increases above the assumed 70 F "as-built" temperature. The concrete primary shield wall analysis as described in Section D.1 below is also based on a 70 F "as built" temperature, but incorporates the thermal responses of the various individual components comprising the reactor support system. Because of the differing approach used in the two analyses, the results should not be construed as being interchangeable. However the conclusions as to the adequacy of these parts of the support system are supported by the analyses performed the differing approaches notwithstanding.

D.1 CONCRETE PRIMARY SHIELD WALL

D.1.a Operating Conditions

The analysis was performed for steady state and transient conditions described in Section C. The steady state condition is the condition that will be attained within the containment following start-up and initial operation of the reactor. The steady state condition incorporates the thermal movements that will occur as a result of a gradual heating of the containment. In other words, the steady state condition is the normal condition.

The transient conditions which are considered all involve a start-up of the reactor after a period of extended cold shutdown (in excess of two weeks) wherein the containment is assumed to have cooled to an ambient temperature of 70 F. Under this condition, maximum temperature differential effects will exist between the steel support system and the primary shield wall. Thus, the transient conditions involve reactor start-up from 70 F to full power, operation at full power with loss of cooling fans and concrete at 70 F. These conditions are analyzed to determine limiting design conditions and are not expected to occur in the actual operation of the reactor.

D.1.b Computer Modeling

To analyze for these various conditions, the interaction of both the steel support system and the associated concrete elements had to be taken into account.

To determine this interaction, a finite element analysis was made using the STARDYNE 3 computer program. Several models were used to determine the effects that temperature change, imposed external load and geometry would have on the total reactor support system. An isometric section of a segment of the reactor internal structure and the corresponding models is shown in Figure 3H-13. The main finite element model was a two (2) dimensional horizontal model (see Figure 3H-14) used for determining the stresses in the reinforced concrete primary shield wall caused by the logos from the steel supports for the reactor as well as pressurization loads. A finite element two dimensional vertical model (see Figure 3H-15) of the primary shield wall and a finite element two dimensional horizontal model of the foundation mat for the primary shield wall (see Figure 3H-16) were developed to determine the spring values and thermal deformations to be used as input data for the main finite element horizontal model to simulate the three (3) dimensional effects.

Vertical Model

Figure 3H-15 shows the vertical finite element model of the lower containment internal structure which provides support to the reactor and other nuclear steam supply system components located in the containment. This model is used to determine the movement due to temperature changes in the mass concrete section from the base mat to the floor at elevation +18.

Temperature distribution through the concrete representing maximum operating conditions is shown in Figure 3H-15. The temperature gradient is based on a straight line gradation between the internal concrete surfaces at 120^0F and the exterior concrete surfaces at 70^0F. For the lower mass concrete section, all exterior surfaces are below grade.

The 120^0F interior temperature is the maximum operating temperature expected; the 70^0F exterior temperature is the soil and/or ground water temperature. Thus, the elements of this vertical finite element model are subjected to an increase in temperature above the "as built" temperature of 70^0F. Expansion will occur in the mass concrete section due to the temperature increase.

The expansive movement will be in a direction radially outward and upward as dictated by the boundary conditions shown in Figure 3H-16. Note that no radial expansion is permitted at the center of the mass concrete from the base mat to the bottom of the cavity (elev -3) because the base is a solid circle. Above the bottom of the cavity, the geometry of the lower mass concrete changes to a thick ring wherein radial expansion is restrained by a so-called "ring action."

To-account for the "ring action" on the radial expansion of the thick ring section, 4 springs were determined at 4 different levels from 4 horizontal models and added to the vertical model (see Figure 3H-16).

The horizontal model used to determine the spring constant is shown in Figure 3H-17. Note that the boundary conditions of the model permit only radial movement. Also, the external forces acting on the modeled segment of the thick ring are an internal radial load applied to the inside face of the cavity at each node point, springs representing the soil reaction on the exterior of the shield building and the "hoop" forces resulting from a radial movement of the segment of the thick ring.

The spring constant value to be used in the vertical model is determined as follows:

The force applied inside the cavity, "P", results in a radial deflection of each node point, "Δ". The spring constant "Kr" is determined by dividing the applied force "P" by the resulting radial deflection "Δ".

Thus, the vertical finite element, incorporating the features discussed above, was used to determine the radial expansion or contraction of the lower mass concrete section due to changes in temperature. The movement affects the primary shield wall and the steel beam-column assemblies supporting the reactor.

Main Horizontal Model

The main horizontal model represents the primary shield wall (above elev +18) and steel beam-column assemblies supporting the reactor. The finite element model is shown in Figure 3H-14. The thermal response of the lower mass concrete supporting the primary shield wall, as determined from the vertical model, is factored into this main horizontal model as follows:

Each node point of the main horizontal model is connected to an "equivalent beam." The "equivalent beam" is proportioned in terms of the structural properties of cross-sectional area, length, modulus of elasticity, and stiffness so as to reflect the response of the lower mass concrete into the nodes of the main horizontal model. The "equivalent beam" provides a means of simulating the effects of the thermal movement of the lower mass concrete, torsional effects within the primary shield wall due to unsymmetrical loads and the restraining effect the lower mass concrete has on the primary shield wall when subjected to pressure loads.

The "equivalent beam" is determined by the following steps:

a) Set the equivalent beam material and beam length:

   Beam material: steel   a = 0.0000065 = (Coef of Expansion)

   E = 29,000 ksi

   Beam length: set it equal to 0.5 or 1 ft in geometry

b) Assume that this equivalent beam can only resist axial and shear forces, no bending or torsional moments can be resisted by this beam. Therefore, all moments of inertia and torsional constant are equal to zero, only cross-sectional areas and beam shear shape factor have to be determined. To include temperature effect, the temperature change in equivalent beam has to be determined.

c) The cross-sectional area of equivalent beam is set so as to reflect the radial spring constant K_r

$$A = \frac{K_r l}{E}$$

where K_r = radial spring constant (see explanation in Section F)

l = assumed beam length

E = 29,000 ksi

d) Shear shape factor of equivalent beam is utilized to represent tangential spring (torsional restraint) on the primary shield wall to resist unsymmetrical loading. It is obtained as follows:

Set

$$K_t = S\frac{GA}{l}$$

where K_t = is tangential spring (see explanation in step F)

G = shear modulus

A = cross-sectional area, from Section C

l = beam length

S = shear shape factor of beam

For steel, $G = \frac{E}{2.5}$

Thus

$$K_t = S\frac{EA}{2.5L} = S\frac{K_r}{2.5}$$

Since K_t and K_r are known, S can be found.

e) Since the beam can be subjected to temperature change, set beam temperature to represent the radial movement, "d", at the main horizontal model due to temperature change in the mass concrete section from the base mat to the floor at Elev +18., i.e., to drop temperature ΔT in beam to represent the pulling-out effect due to temperature increases in bottom mass concrete section.

$$\Delta = \frac{d}{la}$$

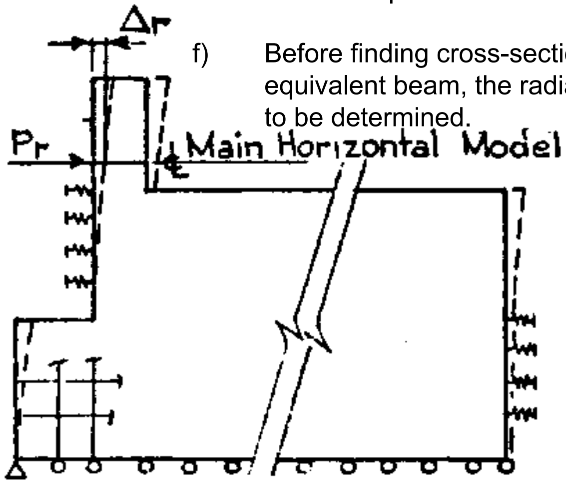
where: Δ = temperature change (or drop) in equivalent beam

l = equivalent beam length

a = 0.0000065

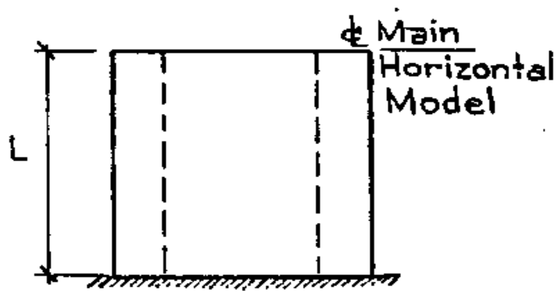
d = is the movement at Main Horizontal Model due to temperature change in mass concrete section from the base mat to the floor elevation +18. (See explanation in Vertical Model.)

f) Before finding cross-section area, shear shape factor and temperature change of equivalent beam, the radial spring constant K_r and tangential spring constant K_t have to be determined.

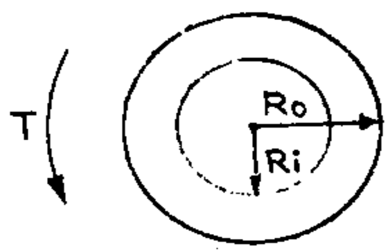


Kr is the radial spring constant which represents the radial movement restraining effect of the lower mass concrete on primary shield wall. By using the vertical model, displacement r in the main horizontal model due to P_r force applied on same location is found by:

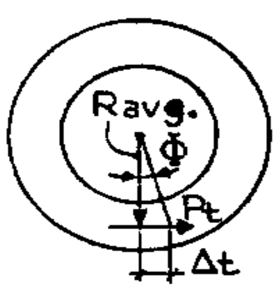
$$K_r = \frac{P_r}{\Delta_r}$$



2a) ELEVATION



2b) TOP VIEW



2c) EQUIVALENT TO TOP VIEW

K_t is the tangential spring constant which represents the torsional restraining effect on primary wall. It is obtained as follows:

$T = P_t R_{avg}$

$\Phi = \frac{T\,L}{J\,G}$

$\Delta = R_{avg}\,\Phi$

$K_t = \frac{P_t}{\Delta_t} = \frac{T}{R_{avg\Phi}} = \frac{J\,G}{R_{avg}^{2}\,L}$

where

T	=	Total resisting torsional moment
P_t	=	Imaginary applied tangential force
Δ_t	=	Tangential displacement due to P_t
L	=	Effective shaft length
J	=	Polar moment of the inertia
G	=	Shear modulus
R_{avg}	=	Average radius in primary shield wall

Stardyne Analysis System

The STARDYNE Analysis System consists of a series of compatible digital computer programs designed to analyze linear elastic structural models. The system encompasses the full range of static and dynamic analysis. These programs provide the analyst with a sophisticated, structural dynamical analysis system.

The STARDYNE system is used to evaluate problems such as the computation of structural deformations and member loads and stresses caused by an arbitrary set of thermal, nodal applied loads and/or prescribed displacements.

The basic concept of the "Finite Element" method is that every structure may be considered as a "mathematical" assemblage of individual structural components or elements. There must be a finite number of such elements, interconnected at a finite number of nodal points. The behavior of this finite element structural model will closely approximate the behavioral characteristics of the real structure.

D.1.c Assumptions

Below is a list of the primary assumptions used in the analysis of the concrete primary shield wall:

a) The analysis assumes all materials remain within their elastic limits, i.e., there is a linear relationship between stresses and strains and plane sections remain plane within the limits of loading.

b) The thermal expansion of the mass concrete section resulting from the heating of the containment is based on a linear distribution of temperature from the internal concrete surfaces to the exterior of the shield building.

c) The maximum allowable principal tension is equal to 7.5 $\sqrt{f'c}$ or 530 psi for the concrete.

d) Unless otherwise stated, a constant modulus of elasticity is assumed throughout the concrete.

e) Axial loads from the steel support beam are transferred into the concrete at the two vertical stiffeners and end plate. No allowance for bond between the steel beam and concrete is made.

f) Unless otherwise noted, no allowance for the effects of creep is made.

g) Only the horizontal hoop reinforcing is included in the model. Vertical and transverse reinforcing is not included.

h) No increase in allowable stresses is considered for short duration loads.

D.1.d Results

The output for the main horizontal model is in terms of forces, moments, deflections, and principal stresses for the nodes or elements of the model. In addition, complete stress plots were developed to show the stress patterns throughout the primary shield wall. The significant forces and deflections for the various loading conditions are shown in Figures 3H-18 through 3H-23. The stress plots are shown in Figures 3H-24 through 3H-28.

For the steady state cases, using the load combinations indicated above, no overstressing of concrete occurred. Also, the maximum lateral movement of the reactor support was within allowable limits.

For the transient conditions, normal operating condition results in stresses within allowable limits. However, load condition (g) of Section C.1 results in tensile stresses in excess of the allowable. Another loss of fans analysis was run reflecting a concrete temperature of 90 F. This case resulted in 3 elements overstressed in tension. This case was rerun with the overstressed elements "cracked"; the results of the "cracked" model showed crack termination. For this condition, no loss of function will result because the load is relieved by additional deformations that will accompany the higher stresses; that is, the load is self relieving. A total deformation of 3/16 inch will relieve the entire load.

Below is a list of items which make the analysis conservative:

a) The maximum containment air temperature, 120 F, was used in design. Lower air temperature will result in less differential thermal stress between steel beams and concrete walls.

b) The concrete strength used in the analysis is the design strength 5000 psi. Actual concrete strength at 28 days averaged 6800 psi with a minimum of 6300 psi.

c) No account has been made in the finite element model for transverse or vertical reinforcing.

d) No allowance has been made for the effects of creep.

e) The 70 F concrete temperature is a limiting design temperature and is not expected to be reached during the life of the plant.

D.2 STEEL BEAM - COLUMN ASSEMBLY

Analysis of the steel portion of the reactor support structural (horizontal girder and vertical column) was performed using the CDC STARDYNE finite element computer program. In this analysis, the structure was idealized in three dimensions using quadrilateral and triangular plate elements capable of carrying both inplane and bending loads (see Figure 3H-29). The ends of the girder were restrained against translations in the vertical and two horizontal directions, and the column base was restrained vertically at anchor bolt locations.

Analysis was performed for the load combinations presented in Section C.2. Each load condition consists of thermal and mechanical loads. Temperature loads were applied to each finite element to represent the thermal load distribution determined by Attachment 1. Concentrated forces were applied at discrete points on the finite element model to represent the mechanical loading in the vertical direction, and in the horizontal direction parallel to the girder.

Element stresses and displacements throughout the structure were determined from this finite element analysis. These element stresses and the vertical displacement at the girder centerline were modified to include the effect of vertical concrete movement, and then compared with the allowable values of stress and displacement. A summary of these comparisons is presented in Table 3H-3. Stresses and displacement do not exceed allowable values as indicated in Table 3H-2.

E. CONCLUSIONS

The reactor support system will withstand the combination of loads postulated in Section C and support deformations are within allowable limits. The overstressing condition identified in the application of load combination 'g' would result in additional concrete deformations. These secondary deformations would reduce the thermally induced loads caused by the heating up of the steel beam-column assembly and would result in movement of the reactor supports of less than 3/32 inches. This would not result in an adverse effect on the primary piping. This condition is considered acceptable from a safety standpoint. Therefore, the reactor support system can accept all design loading conditions postulated.

F. THERMAL SHIELD REMOVAL AND CORE SUPPORT BARREL REPAIR EFFECTS

The removal in 1984 of the reactor thermal shield from the St. Lucie 1 reactor vessel will produce an increase in the radiation level in the reactor cavity area, and consequently in the temperature. Temperature increases in both the primary shield wall concrete and the steel reactor support structure will be experienced.

Extensive analysis have been performed to determine the effects of various thermal loading conditions in combination with the other loads imposed on the reactor support structure. It was recognized that the removal of the reactor thermal shield necessitated a reinvestigation of some of these previously evaluated cases in order to demonstrate that the earlier conclusions as presented in Sections "A" through "E" are not altered by the present modification.

The earlier analysis had indicated that two loading cases were particularly critical from the structural point of view. The LOCA condition, presented as combination C.1.a(d), when later modified by the addition of the North Anna Syndrome loads developed by Combustion Engineering (CE) represented the worst loading case for the concrete primary shield wall structure. The appropriate temperature distribution for this condition is the normal operating steady state condition. This loading case also resulted in some local yielding of the steel reactor support girder, however, this was considered minor and not warranting re-evaluation for the present investigation.

What was considered critical for the steel girder-column assembly was the vertical deformation, which could potentially impact on the CE stress analysis for the reactor coolant piping. This parameter was clearly sensitive to changes in the thermal distribution. The appropriate loading combination was the loss of fans condition, presented as combination C.2.a(d). The minimum required operating fans consist of a sufficient number of containment fan coolers to maintain containment atmosphere temperature $\leq 120^{o}$F, 1 of 2 reactor cavity fans, 1 of 2 CEDM cooling fans and 1 of 2 reactor support cooling fans. That condition was later modified and re-evaluated by allowing the time for operator action following fan failure to increase from 15 minutes to 45 minutes. If the minimum required operating fans can not be restored within 45 minutes, operator action to trip the reactor is required. It is this latter case that was considered in the present analysis.

This analysis presents the evaluation of two cases: (1) the combination of the new thermal loads with the previously evaluated North Anna Syndrome loads and load, equipment, subcompartment pressure and seismic loads; (2) the vertical growth of the steel girder-column assembly under thermal conditions generated by a loss of fans in combination with the new radiation heating conditions.

F.1 ENERGY DEPOSITION RATE

Removal of the reactor thermal shield increases the radiation level in the reactor cavity area, and consequently, the temperature. CE calculated the new energy deposition rate radially outward from the reactor vessel in the reactor cavity, and a partial distance into the primary shield wall concrete at the elevation of the reactor core centerline. A deposition rate in the reactor vessel support column was also determined.

Ebasco estimated the vertical energy deposition rate profile in the support column based on measured Unit 1 neutron flux data in the cavity. Assuming the energy deposition rate will fall off in the same manner as the measured fast neutron flux moving up or down from the core centerline, the approximation is made that the deposition rate will decrease from $1.3x\ 10^{-3}$ w/cm^3 in the support column at the centerline, to 2.9×10^{-4} w/cm^3 at the top and bottom of the core (a factor of 4.5 decrease). These energy deposition rates became source terms for the thermal analysis portion of the study.

F.2 THERMAL ANALYSIS

The effect of the enhanced radiation heat load on temperatures in the reactor vessel supports and the primary shield wall is examined using several simplified heat transfer models. The results show a peak temperature in the primary shield wall concrete of 148^0F at the elevation of the core midplane and about one-half foot into the wall, reflecting the influence of the input radiation heat load. Similarly the vertical reactor vessel support leg shows a peak temperature of 141^0F at the elevation of the core midplane, falling to a steady 124^0F below the core. Above the core midplane a similar temperature gradient is seen until the elevations where the influence of the reactor leg is felt. As the top of the horizontal vessel support is just above the top of the core the temperature over the previous detailed transient heat transfer studies was used as the basis of the temperature distribution for the transient conditions examined in the structural analysis.

Two models were constructed for analysis by the heat transfer code HEATING5. The heat load derived for the CE data was placed in: (1) a two dimensional model of a section of the primary shield wall; and, (2) a model of the vertical support leg and concrete below the leg. It was seen in the previous heat transfer studies that the influence of the RV leg on the support was restricted to the area immediately around the interface between the support and the shoe. The elevation dependence of the heat load will then impose the greatest loads below this elevation, within the vertical support leg.

To study this effect a detailed model of the vertical support leg was constructed and the heat load imposed as a function of elevation. Convective heat transfer to 120^0F air was assumed on exposed surfaces with a coefficient of 0.85 BTU/hr-ft^2-^{0}F. Radiation to a 150^0F surface (either the RV insulation or the surface of the now-heated primary shield wall) was assumed on the appropriate surfaces. Steady-state runs were made with these models for these conditions.

For the vertical support leg model, Figure 3H-30 shows the centerline temperature as a function of distance from the top to the bottom of the support leg. The distribution shows the effect of the heat load, with the peak temperature occurring at the centerline elevation of the core, and failing off symmetrically in both directions from the centerline to a temperature of 124^{O}F below the core and in the base concrete. This result suggests that the influence of the additional heat load on the horizontal support will be to raise the temperature in the exposed steel from 5 to 10^{O}F in the high temperature region below the interface with the support foot.

These results were used as the basis for the study of the 45 min loss-of fans transient case. The increase in temperature seen in the supports and shield wall between the present study and the previous analysis as applied as an increase on top of the transient conditions, as the heat load from the core will not change during most of the transient. The change of the thermal gradient will change the rate of temperature increase in the leg and horizontal support by limiting heat flow from the support foot; on the other hand, temperatures will slightly increase when forced convection ceases and turbulent natural convection from the exposed surfaces begins. Increases in temperature over the original analysis are shown in Figure 3H-31.

F-3 DEVELOPMENT OF STRUCTURE MODEL

The temperature distribution in the steel girder-column assembly and the concrete primary shield wall were developed as described above for the normal operating steady state condition. The temperature gradient in the mass concrete was then determined assuming a straight line gradient between the primary shield wall temperatures and the exterior concrete surfaces (assumed at 70^{O}F).

Vertical and horizontal models of the reactor cavity as previously described were used for analysis. The EBS/NASTRAN cracked element program, developed by Ebasco and linked to the commercially available NASTRAN program, was used.

Using the developed temperature distribution in the vertical model (Figure 3H-32), the restraining effect due to structural stiffness and vertical growth of the lower mass concrete on the two dimensional horizontal model (Figure 3H-33) was obtained. This effect was represented in the horizontal model by a temperature drop in the fictitious radial beams located around the outer periphery of the primary shield wall. (The effective "spring constants" for these radial beams had been determined by the application of a range of horizontal loads to the vertical model at the elevation of the steel girder.)

The temperatures and LOCA loads (including pressure loading on the cavity wall) were applied to the horizontal model. The resulting total radial horizontal forces in each sector were obtained and proportioned for a 17^{O} arc, later to be used for the 17^{O} arc vertical model. An iterative process was used whereby these resulting radial forces from the horizontal model run were then used to redetermine the radial stiffness or "spring constants" for the fictitious radial beams until the actual forces produced by the horizontal model were reasonably in agreement with the forces upon which the radial stiffness were based. Table 3H-4 shows a comparison of assumed versus actual loads for each sector for the final iteration.

F-4 STRUCTURAL ANALYSIS-PRIMARY SHIELD WALL

NASTRAN run # HKGAJLD dated March 14, 1984 for the horizontal model determined the response of the primary shield wall to all the imposed loads. Figures 3H-34 through 3H-36 show the cracking pattern in sector I, II & III where the steel beams are embedded. (Cracked elements are shown shaded.)

NASTRAN run # VKGAJDA dated March 16, 1984 determined the response of the vertical model to the forces derived from the horizontal model run. No other forces were applied since these were already incorporated in the horizontal model run. Cracking patterns for each sector are shown in Figures 3H-37 through 3H-42.

F.5 VERTICAL GROWTH ANALYSIS

A separate analysis of the steel girder-column assembly was performed to determine whether the vertical growth of the structure under the new radiation heating condition in combination with a loss of cavity and support cooling would still be within the 3/16 inch allowable originally established by Combustion Engineering for differential movement between the reactor support and the steam generator sliding base support. This analysis conservatively assumes that operator action to trip the reactor is not initiated until 45 minutes after loss of cavity cooling.

The temperature distribution developed for the girder column assembly was derived only for the normal operating condition. It was therefore, necessary to extrapolate this information to obtain a temperature distribution applicable to the case of present concern, i.e., the 45 minute shutdown loss of fans case. The following procedure was used:

1. The new temperature distribution in the girder-column assembly (Figure 3H-43) was compared with that for the original normal operating condition, represented in Figure 3H-A12.

2. At a number of points along the column and girder, an incremental temperature increase was determined by subtracting the original temperature from the corresponding new temperature.

3. The incremental temperature increases were added to the corresponding temperatures for the 45 minute shutdown loss of fans case to obtain a new temperature distribution representing the condition of present concern (See Figure 3H-31).

Vertical displacement was then determined by a ratio of weighted average temperature rises between the present case and an earlier case of known vertical displacements. To this value were added the vertical displacements due to the equipment load on the girder and due to the thermal movement of the concrete at the girder ends and the column base.

Finally, the vertical displacements of the mass concrete and concrete pedestal under the steam generator sliding base support were subtracted to obtain the desired differential vertical displacement.

F.6 CONCLUSION

The maximum horizontal load on the primary shield wall is carried by sector III of the horizontal model. The maximum tensile stress in the vertical reinforcing steel is 27.6 ksi, which is within the allowable of 0.9 x 40ksi.

The differential vertical displacement of the reactor support girder relative to the top of the steam generator pedestal is 0.184", which is within the allowable of 3/16."

The foregoing analysis therefore demonstrates that the removal of the reactor vessel thermal shield does not alter the conclusions presented in Sections "A" through "E".

REFERENCES TO APPENDIX 3H

1. Letter R. E. Uhrig (FPL) to D. K. Davis (NRC) Re: St. Lucie Unit 1, Docket 50-335, Reactor Pressure Vessel Support System, L-77-265 dated 8/30/77.

TABLE 3H – 1

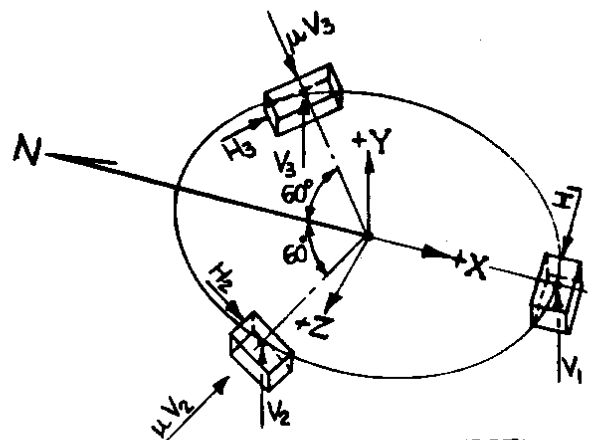
ST. LUCIE PLANT – UNIT 1

REACTOR VESSEL SUPPORT LOADS ($X10^6$ 1b)

	Normal Operating		OBE Seismic			DBE Seismic			Pipe Accident (1)		
Condition Load	Dead Weight	Thermal +D. WT	±X	±Y	±Z	±X	±Y	±Z	A*	B*	C*
H1	0	.028	.005	.002	-.644	.011	.005	-1.288	.056	.024	.024
V1	.666	1.155	.032	.335	-.046	.064	.670	-.092	.109	1.293	1.293
$_{\mu}$V1	± .195	± .350									
H2	0	-.091	1.226	.001	+.355	2.452	.003	+.710	-6.370	-2.450	-2.332
					-.264			-.528			
V2	.634	.726	.017	.253	.380	.035	.507	.761	0	5.040	1.028
$_{\mu}$V2	±.195	±.215									
H3	0	.079	1.139	-.019	.270	2.278	-.038	.540	-6.270	-2.332	-2.450
				-.060	-.349		-.120	.698			
V3	.634	.741	.367	.623	.743	-.006	.506	.746	0	1.028	5.040
$_{\mu}$V3	±.195	±.215									

*A – South Hot Leg Guillotine Break including DBE Seismic (includes 15% increase)
B – 1 A Cold Leg Vertical Slot Break including DBE Seismic
C – 1B Cold Leg Vertical Slot Break including DBE Seismic

Note 1: Pipe accident loads include a dynamic factor of 2, normal operating loads and maximum seismic loads. The direction of the seismic forces included in each accident load was chosen to result in the largest load at each component. This table only lists those accidents which cause critical design loads at one or more support points (cold leg guillotine is not limiting).



REACTOR VESSEL SUPPORT REACTIONS

TABLE 3H-2

Allowable Stress and Displacement Criteria - Steel Beam-Column Assembly

Element Stress versus Allowable Stress,

- Normal operating conditions*

flexure = 0.6 y
shear = 0.4 y

- Accident conditions**

flexure = 0.96 y
shear = 0.96 y/√ 3

Girder Centerline Vertical Displacement versus Allowable Displacement,

girder centerline = 3/16"

* Allowable stresses for normal operating condition are obtained from Section 1.5 of the AISC Code.

**Allowable stresses for accident conditions are based on a 60% increase over normal operating allowables as permitted by Enclosure 2 to AEC ltr to FPL of Aug 29, 1973.

TABLE 3H-3

Load Condition	Maximum Stress, ksi		Maximum Relative Vertical Displacement At Girder Centerline (Inch)	
	Transient Stage	Steady State Stage	Transient Stage	Steady State Stage
Normal Operation	-20.0 (<25.7)	-18.0 (<25.7)	0.065	0.056
LOCA	-36.0 (<41.2)	-39.5 (<41.2)	0.092	0.083
Breakdown (Fan Failure)	-33.8 (<41.2)	-30.5 (<41.2)	0.181	0.172
Post Loca (Post Breakdown)	-33.8 (<41.2)	-30.5 (<41.2)	0.181	0.172
			< 0.188 inch	

Note: Shear stresses for above load conditions are less than 70% of allowable shear stress.

TABLE 3H-4

COMPARISON OF LOAD UPON WHICH RADIAL STIFFNESSES WERE BASED VERSUS HORIZONTAL MODEL FORCES

SECTOR	"LOAD" DETERMINING STIFFNESS (KIPS)	"FORCE" ACTUALLY TRANSMITTED (KIPS)
I	611	592
I-II	466	451
II	405	390
II-III	1240	1254
III	1590	1637
IV-I	1104	1083

ALL LOADS ARE PROPORTIONED FOR USE IN THE 17^0 ARC VERTICAL MODEL

REFERENCE: COMPUTER RUN #HKGAJLD; DT. 3/14/84

Security Related Information
Figure Withheld Under 10 CFR 2.390

Am. 3-7/85

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY CUTAWAY ISOMETRIC

FIGURE 3H-1

Security Related Information
Figure Withheld Under 10 CFR 2.390

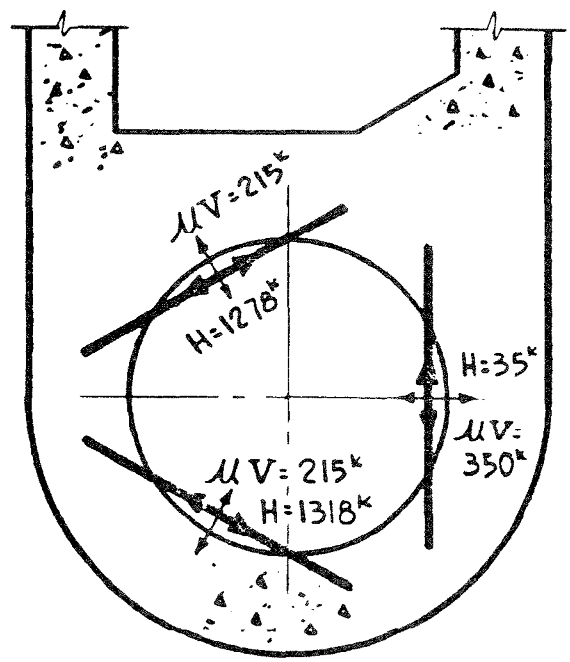
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY PLAN AND SECTIONS

FIGURE 3H-2

A. NORMAL OPERATION WITH OPERATING BASIS EARTHQUAKE

$W = D + L^1 + A + OBE + T_1$



PLAN

NOTE: LOADS SHOWN ARE LOADS TRANSMITTED FROM REACTOR TO REACTOR SUPPORTS.

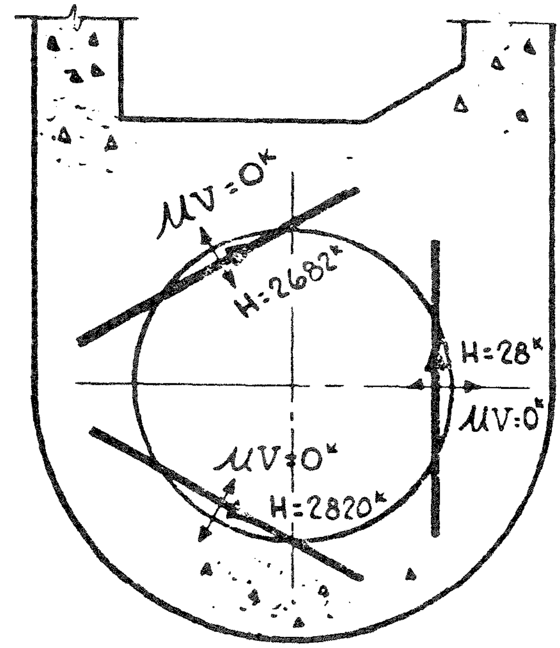
FLORIDA POWER & LIGHT COMPANY **ST. LUCIE PLANT**
REACTOR SUPPORT LOADING DIAGRAM – SHEET 1 FIGURE 3H-3

N

B. LOCA CASES:

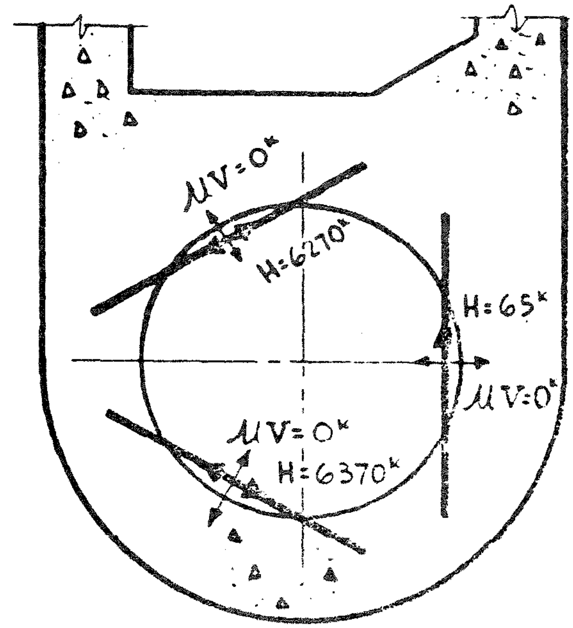
$W = D + L^1 + A + P + Q + DBE + T_1$

1. COLD LEG SLOT



PLAN

2. NORTH OR SOUTH HOT LEG GUILLOTINE



PLAN

NOTE: LOADS SHOWN ARE LOADS TRANSMITTED FROM REACTOR TO REACTOR SUPPORTS.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR SUPPORT LOADING DIAGRAM – SHEET 2

FIGURE 3H-4

N

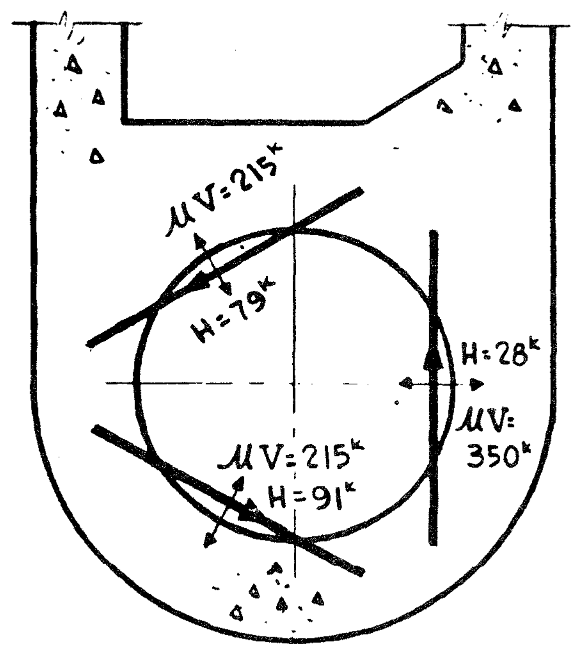
C. LOSS OF FANS:

$W = D + L^1 + A + T_4$

D. POST LOCA CASE:

$W = D + L^1 + A + T_3$

LOADS SAME AS LOSS OF FANS.



PLAN

NOTE: LOADS SHOWN ARE LOADS TRANSMITTED FROM REACTOR TO REACTOR SUPPORTS.

LEGEND:

D	DEAD LOAD
L^1	EQUIPMENT DEAD LOAD
P	LOSS OF COOLANT ACCIDENT PRESSURE LOAD
Q	LOSS OF COOLANT ACCIDENT PIPE OR EQUIPMENT LOAD
A	NORMAL EQUIPMENT OR PIPE ANCHOR LOAD
T_1	NORMAL OPERATING THERMAL LOAD
OBE	OPERATING BASIS EARTHQUAKE LOADS
DBE	DESIGN BASIS EARTHQUAKE LOADS
T_3	POST LOCA THERMAL LOADS
T_4	LOSS OF FANS THERMAL LOADS

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR SUPPORT LOADING DIAGRAM – SHEET 3

FIGURE 3H-5

Security Related Information
Figure Withheld Under 10 CFR 2.390

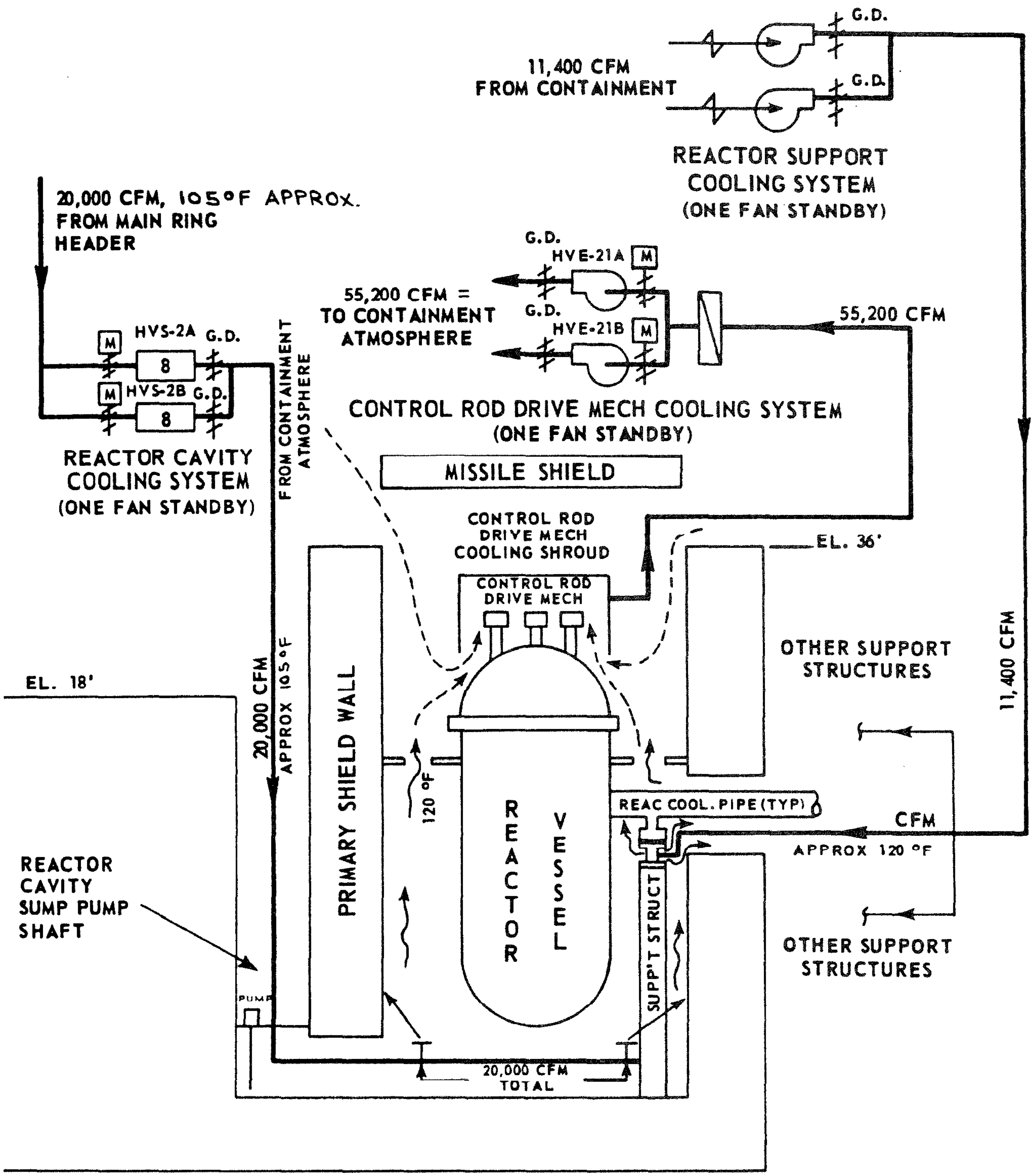
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR BUILDING INTERNAL
CONCRETE – PLANS AND SECTIONS
MASONRY – SHEET 1
FIGURE 3H-6

Security Related Information
Figure Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR BUILDING INTERNAL
CONCRETE – PLANS AND SECTIONS
MASONRY – SHEET 2
FIGURE 3H-7



Note

CEDM cooling based on 14 x 14 fuel assembly requirements. Compatibility with 16 x 16 design will be evaluated.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

AIR FLOW DIAGRAM FOR REACTOR CAVITY, SUPPORT LEG AND CONTROL ROD DRIVE MECHANISM COOLING SYSTEMS

FIGURE 3H-8

Security Related Information
Figure Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY REINFORCING
SHEET 1

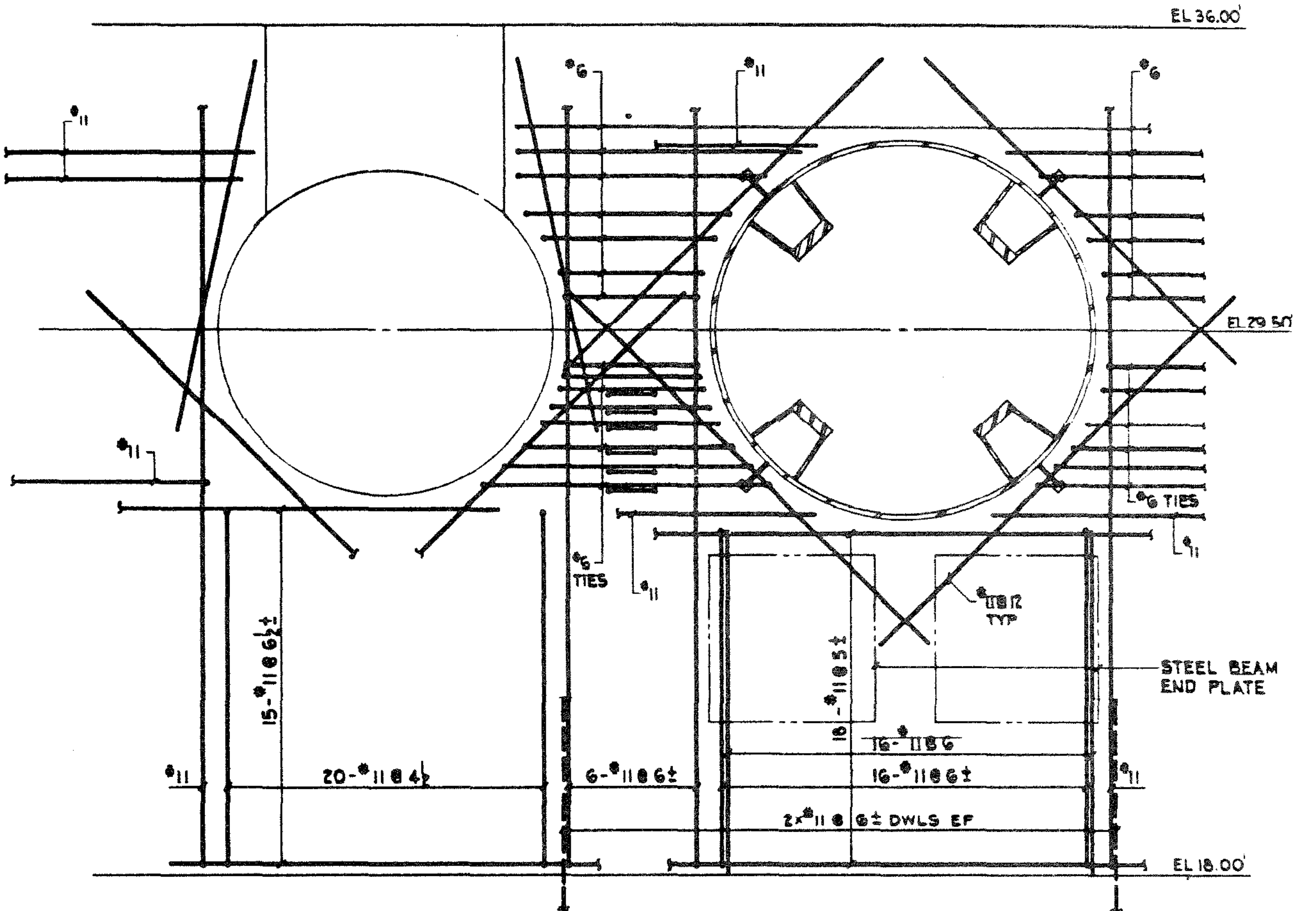
FIGURE 3H-9

Security Related Information
Figure Withheld Under 10 CFR 2.390

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY REINFORCING
SHEET 2

FIGURE 3H-10

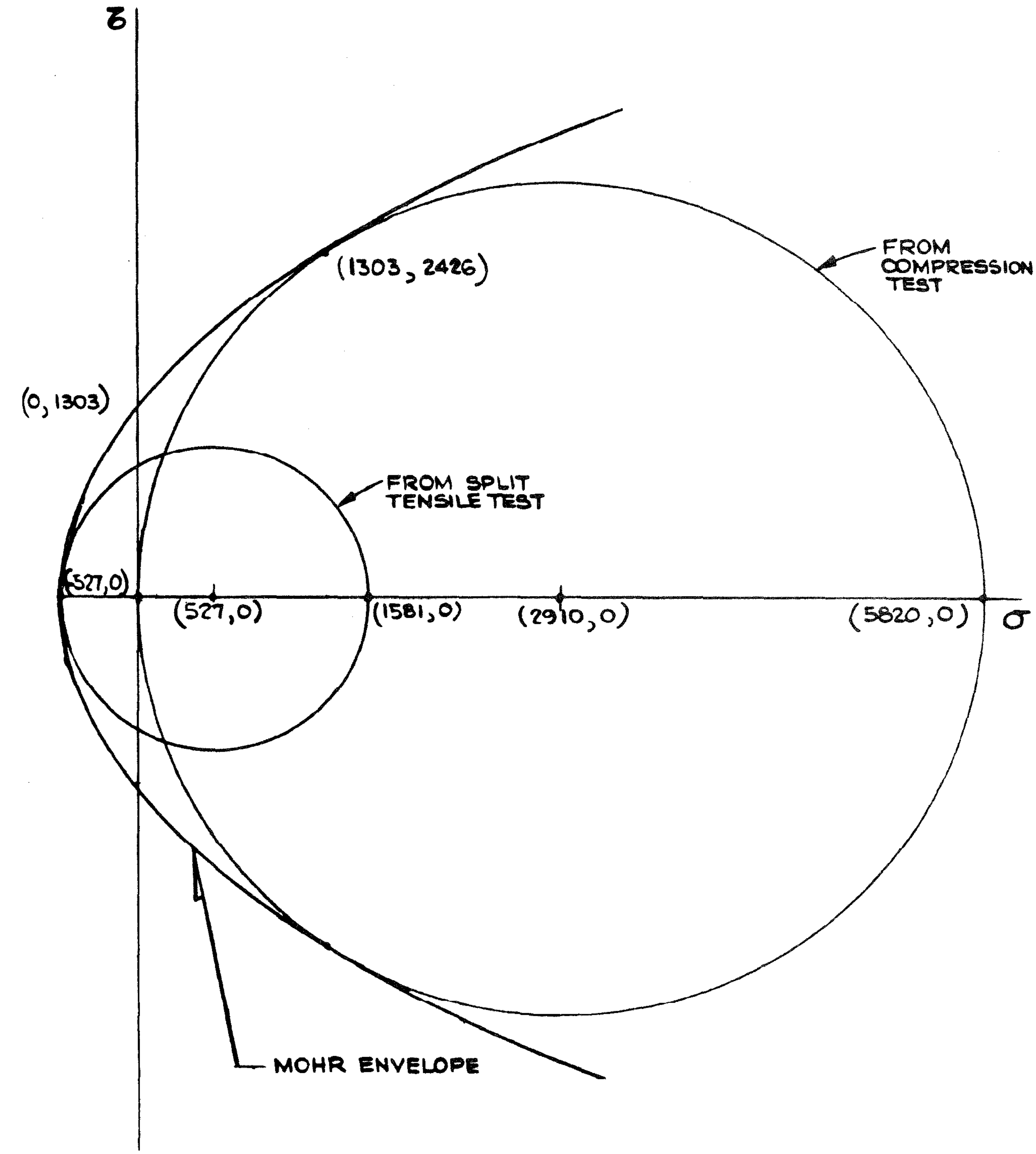


SECTION B-B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY REINFORCING
SHEET 3

FIGURE 3H-11

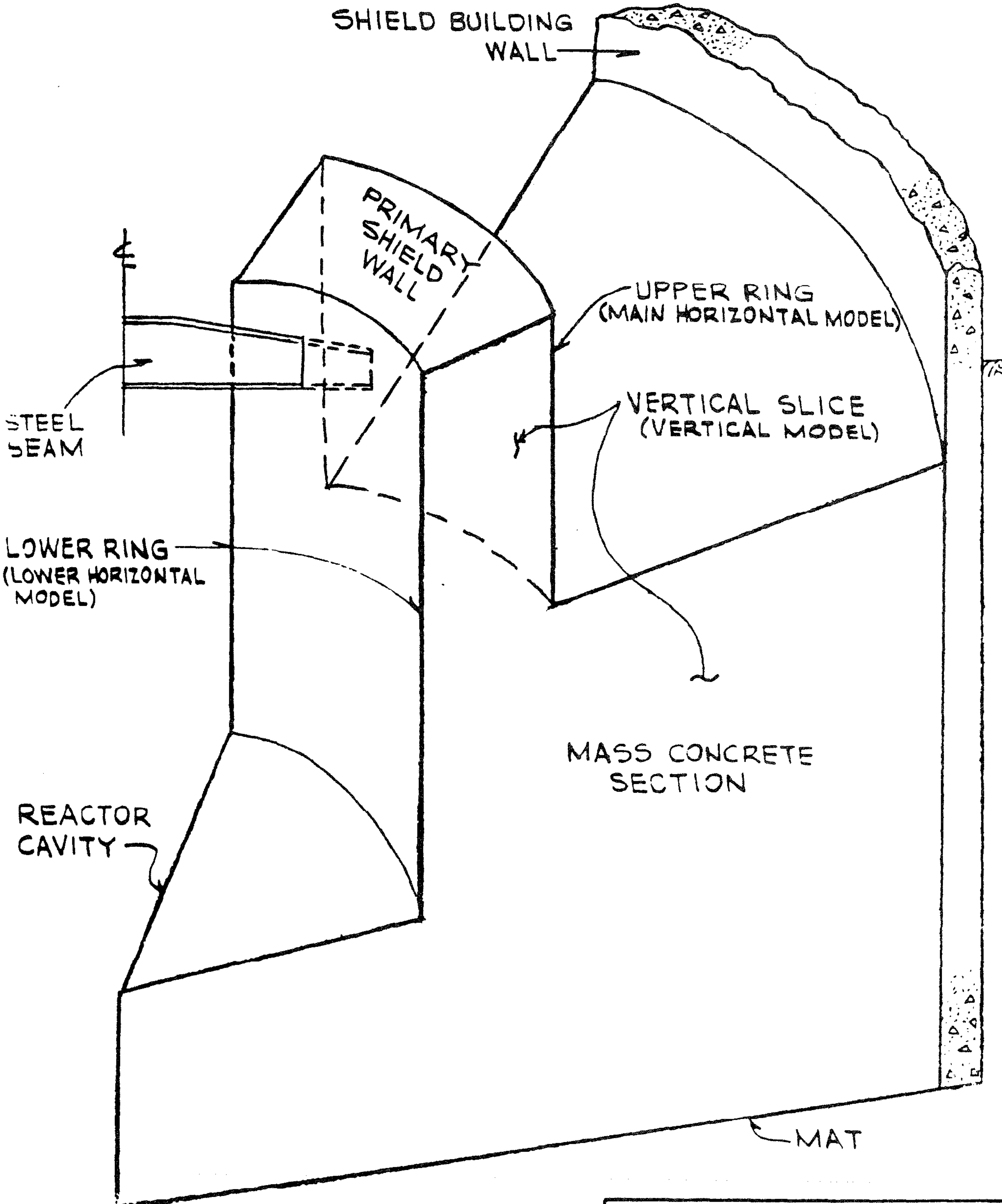


UNITS PSI

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

MOHR'S FAILURE ENVELOPE
PLAIN CONCRETE

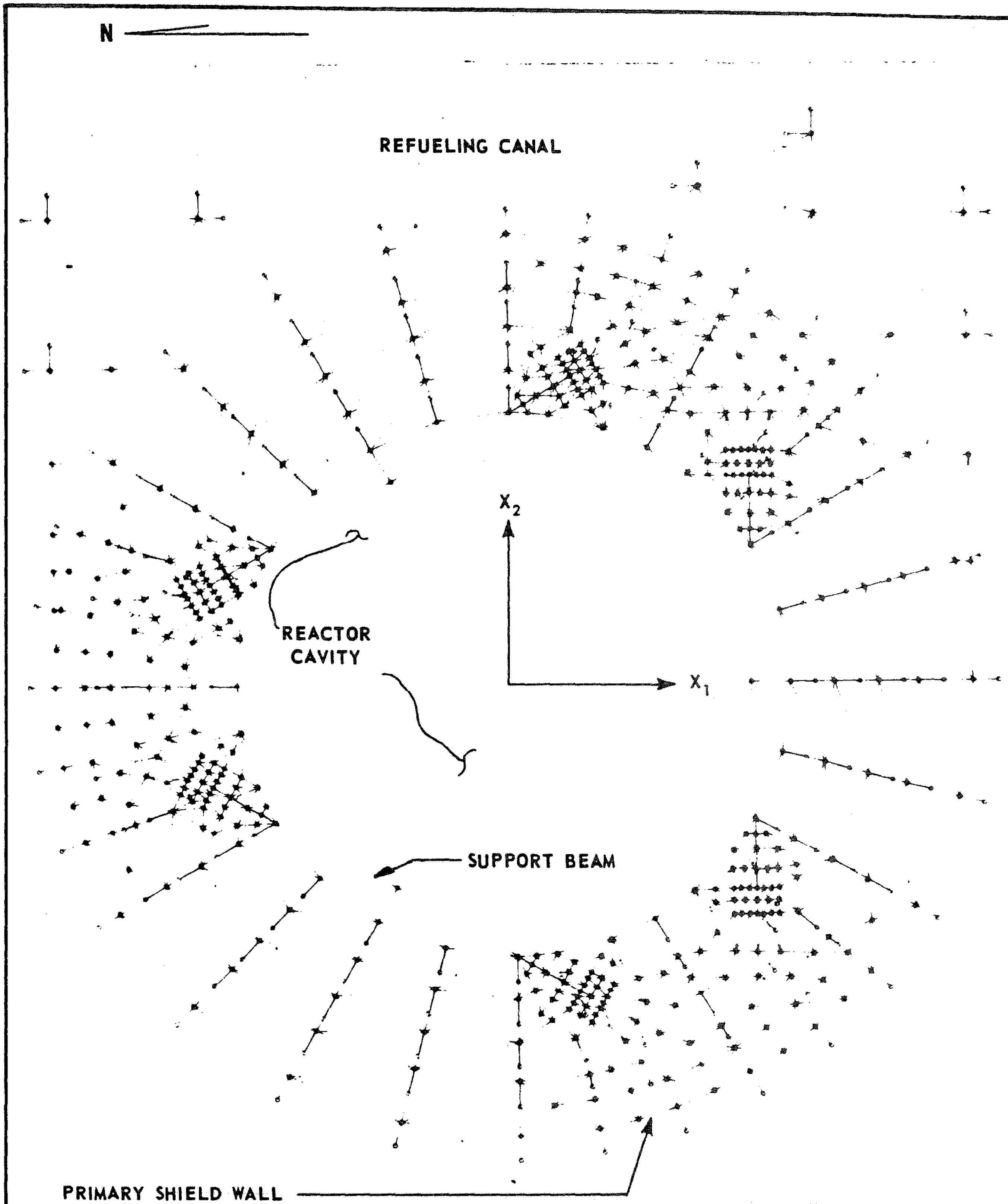
FIGURE 3H-12



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR INTERNAL STRUCTURE
ISOMETRIC SECTION

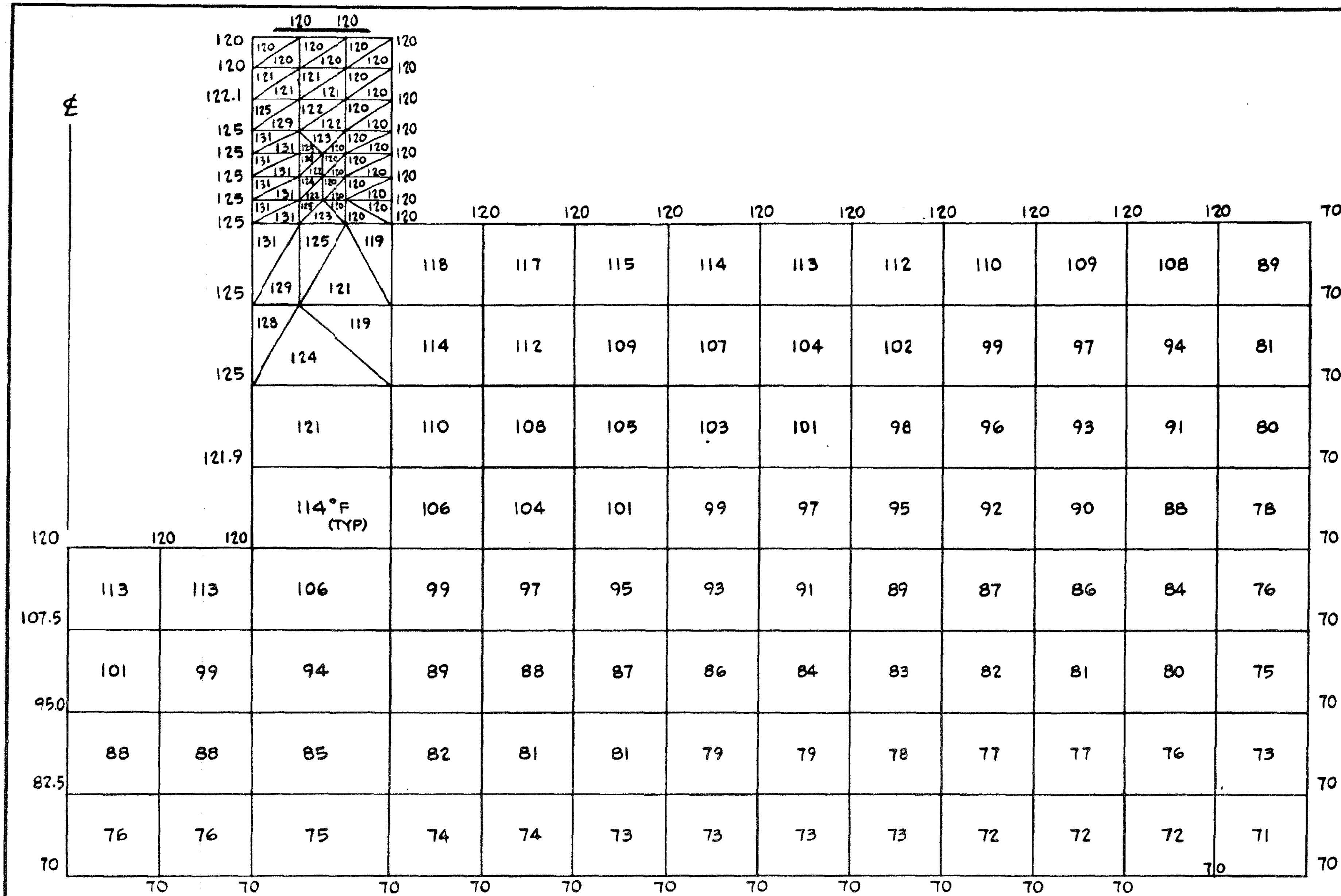
FIGURE 3H-13



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR SUPPORT
MAIN HORIZONTAL MODEL

FIGURE 3H-14



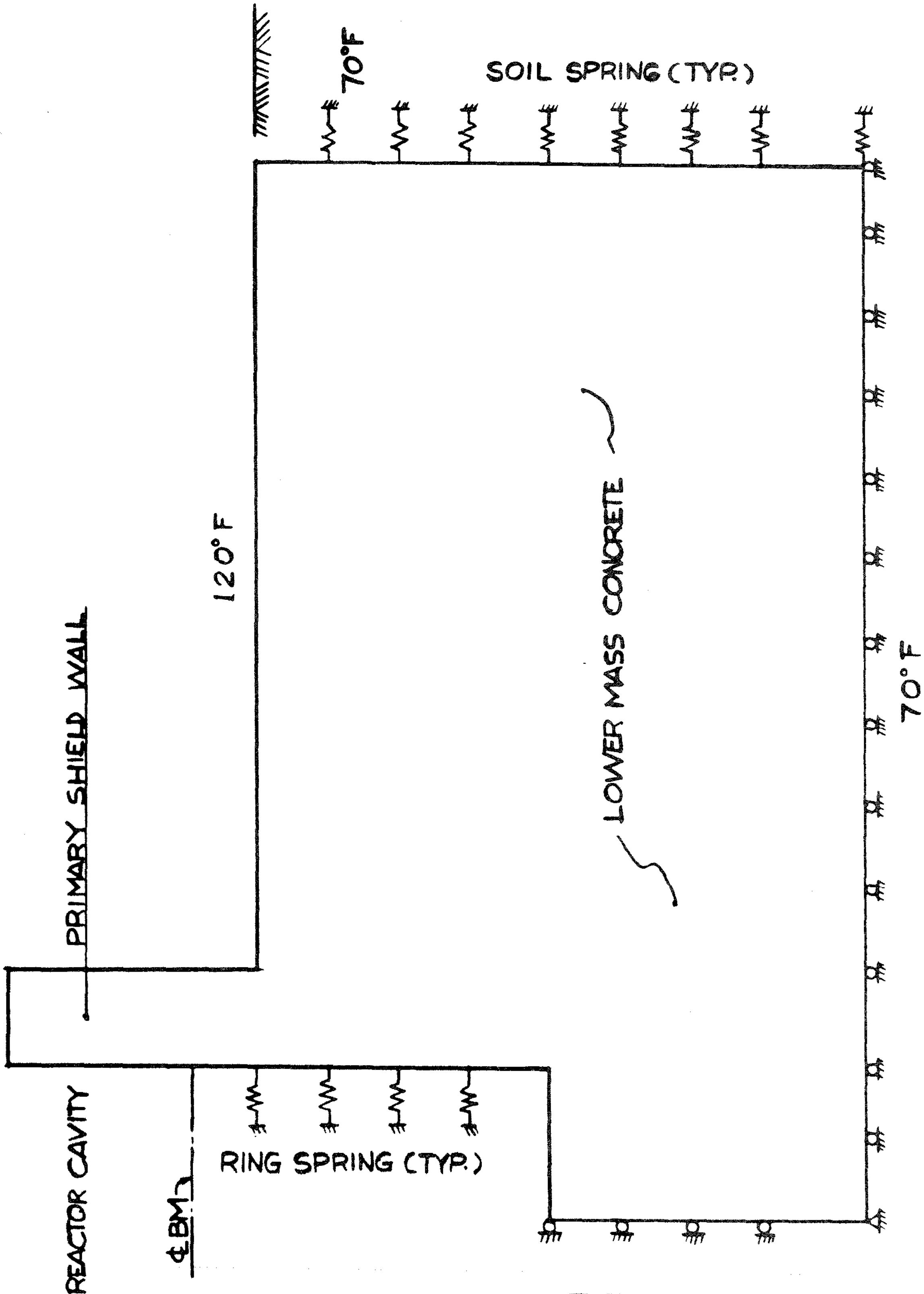
VERTICAL FINITE ELEMENT MODEL WITH NORMAL TEMPERATURE DISTRIBUTION

FLORIDA POWER & LIGHT COMPANY

ST. LUCIE PLANT

VERTICAL FINITE ELEMENT MODEL

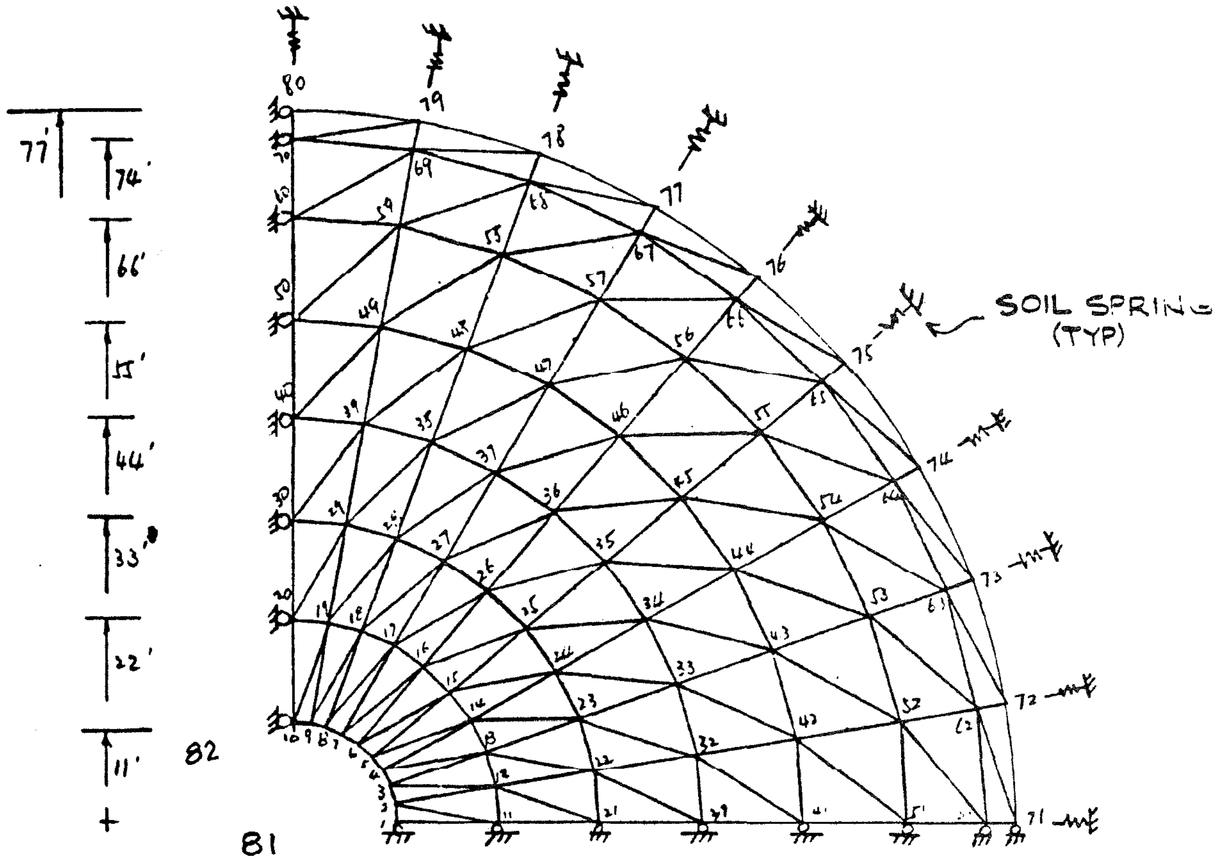
FIGURE 3H-15



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

LOWER MASS CONCRETE
BOUNDARY CONDITIONS

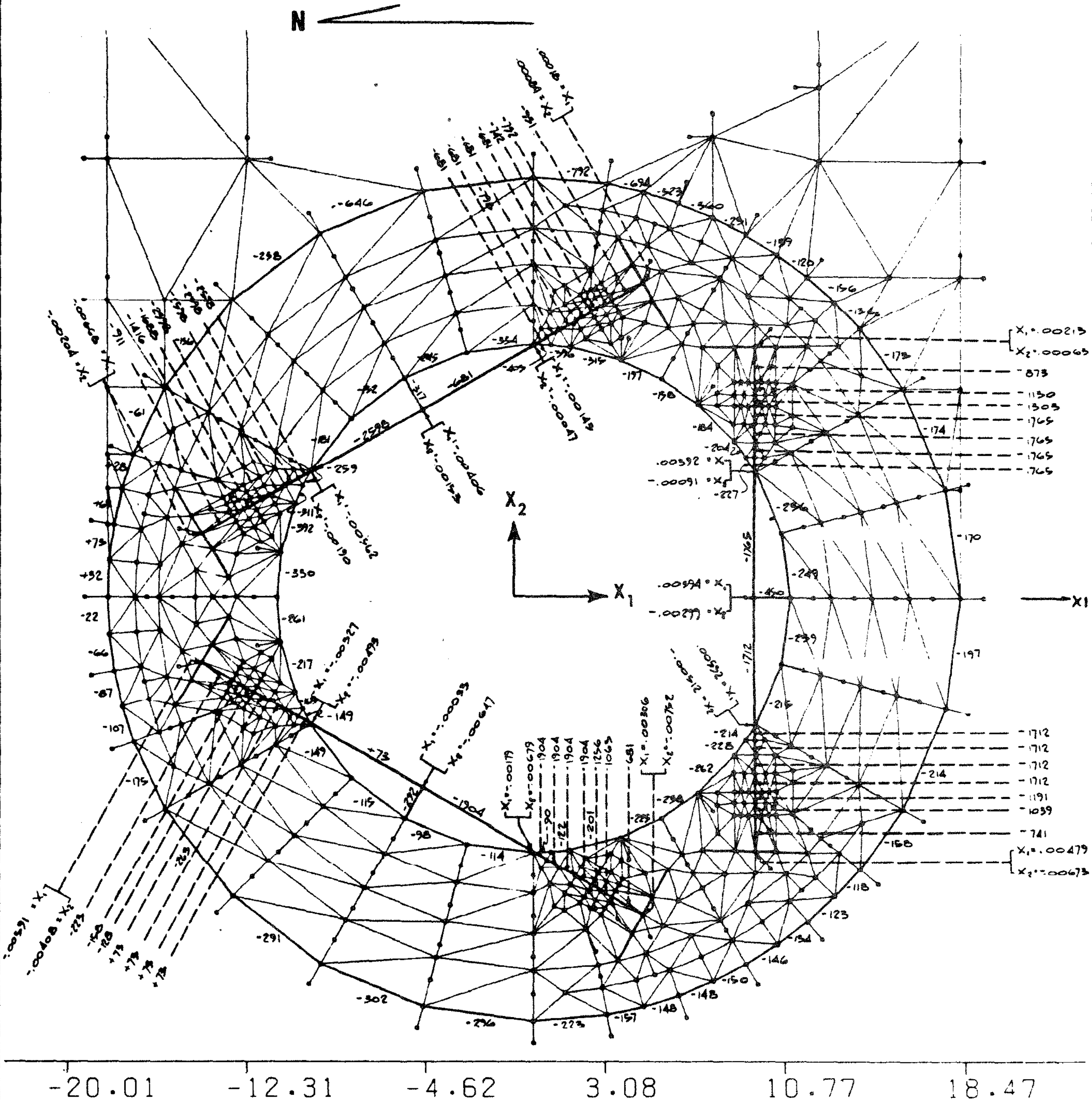
FIGURE 3H-16



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

PLAN-HORIZONTAL FINITE ELEMENT MODEL FOR LOWER RING

FIGURE 3H-17



NORMAL OPERATING – STEADY STATE

U = 1.5 (D + L' + A + T1)

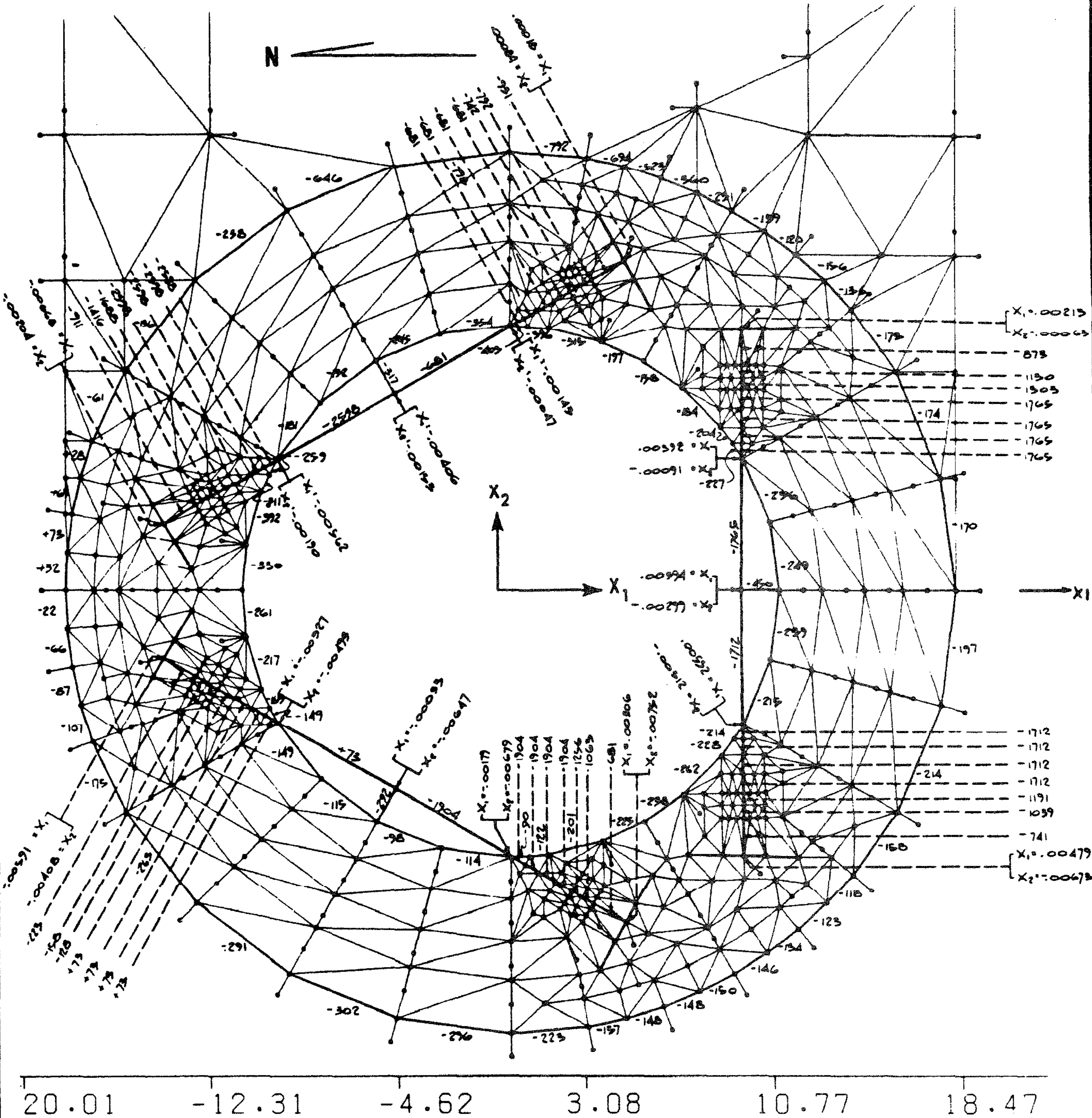
UNITS: DISPLACEMENT IN FEET
FORCES IN KIPS

NOTES: + TENSION
\- COMPRESSION

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
DISPLACEMENTS AND FORCES – SH. 1

FIGURE 3H-18



NORMAL OPERATING WITH OPERATING BASIS EARTHQUAKE

U = 1.5 (D + L' + A + T1) + 1.5 (OBE)

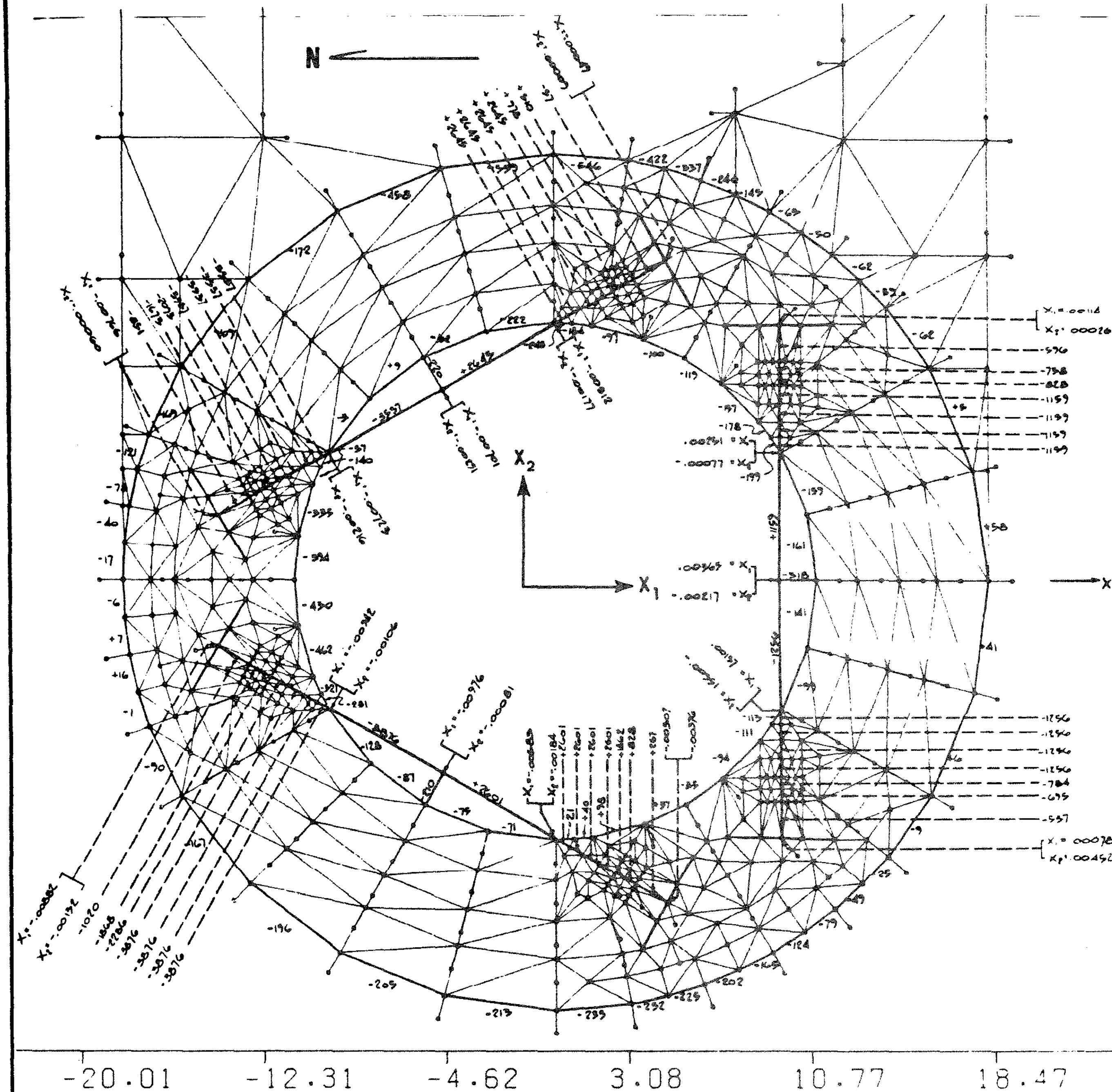
CONCRETE TEMP = 120°F

UNITS: DISPLACEMENT IN FEET
FORCE IN KIPS

NOTES: + TENSION
- COMPRESSION

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
DISPLACEMENT AND FORCES
SHEET 6
FIGURE 3H-19



LOCA – STEADY STATE

$U = T1 + 1.15\,(D + L' + A + P + Q + DBE)$

UNITS: FORCES IN KIPS
DISPLACMENTS IN FEET

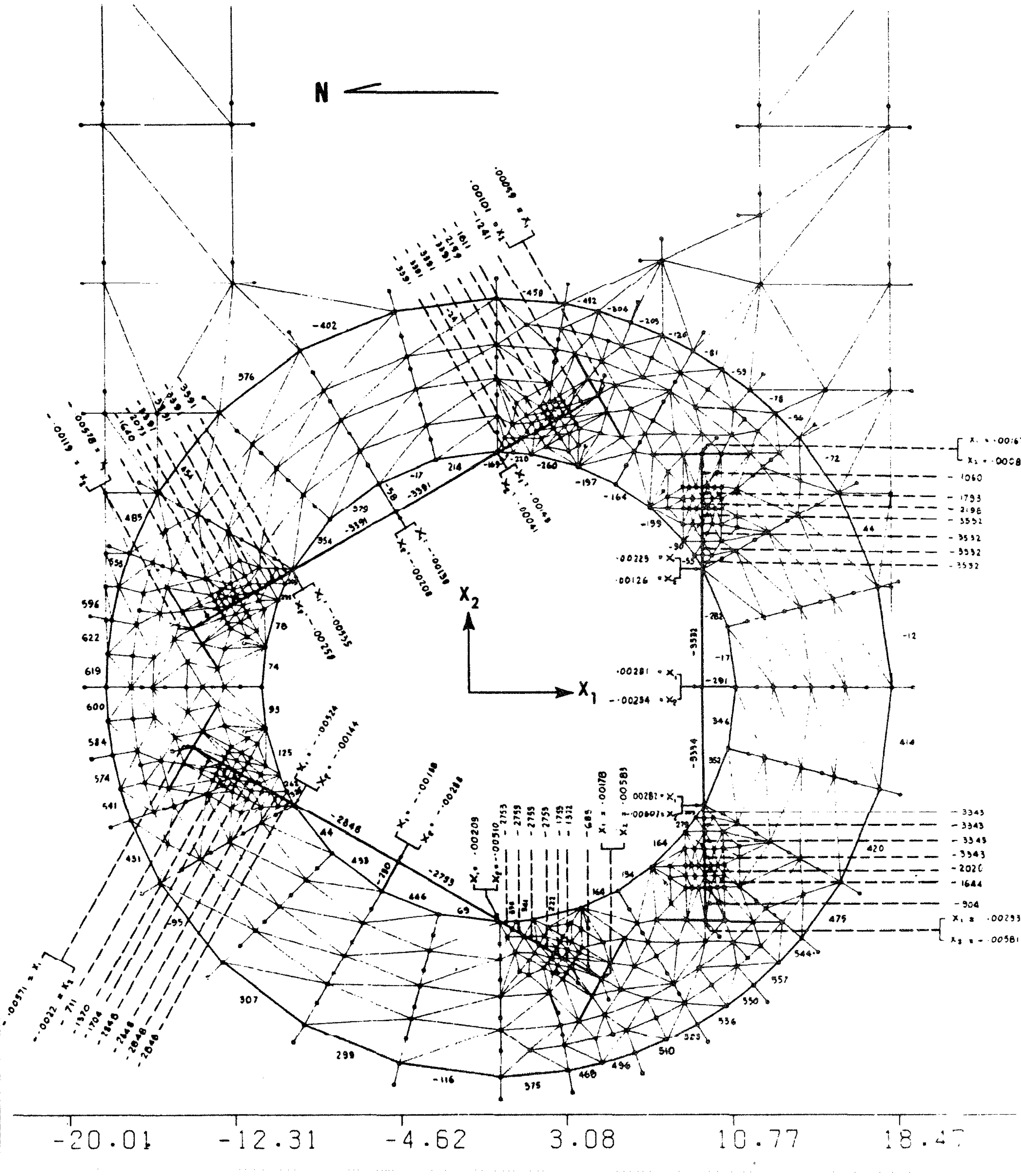
NOTES: + TENSION
– COMPRESSION

LOCA – SOUTH HOT LEG GUILLOTINE BREAK
(CRITICAL LOCA LOADING)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
DISPLACEMENT AND FORCES – SHEET 5

FIGURE 3H-20



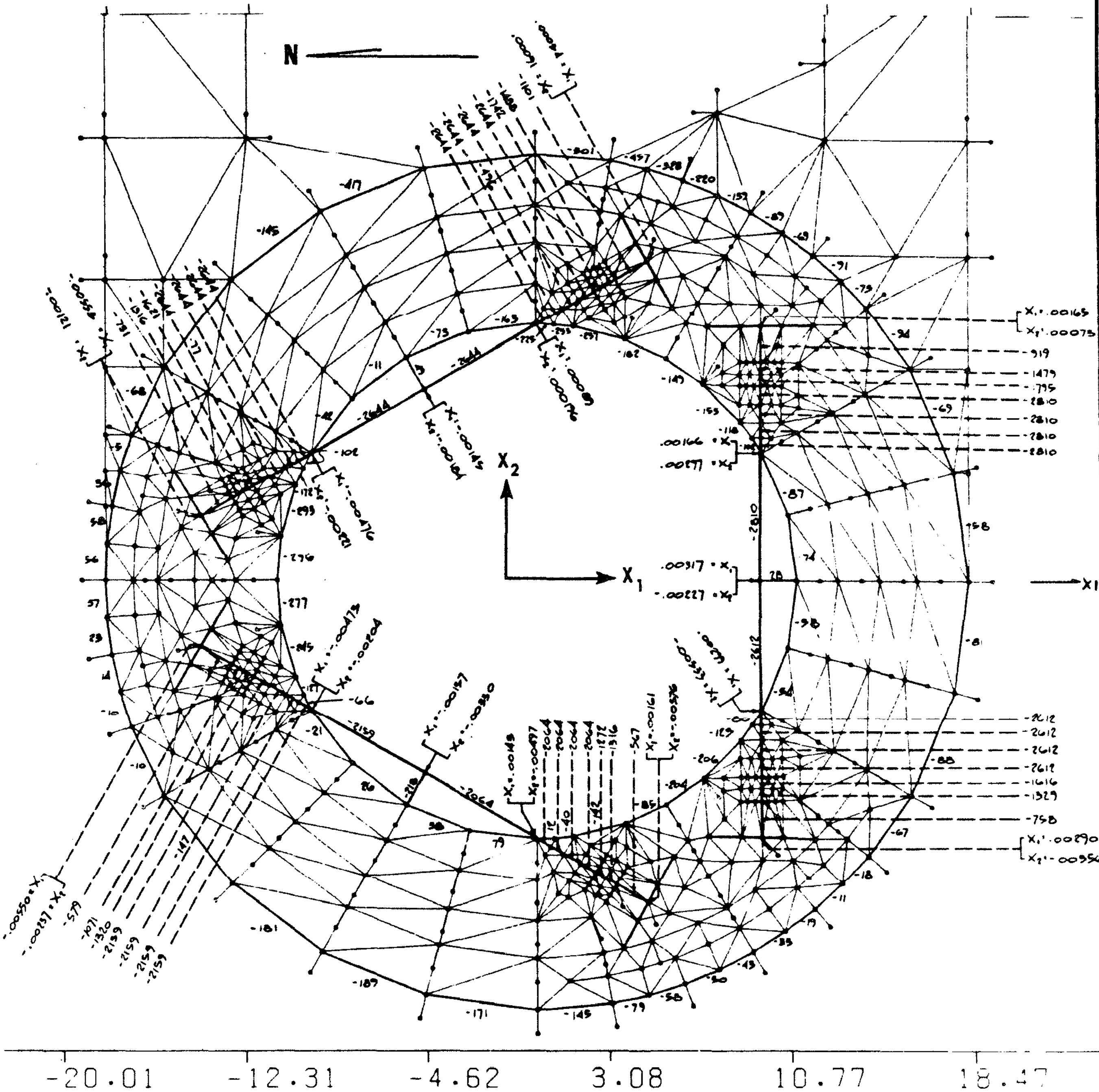
POST LOCA – STEADY STATE
U = D + L' + A + T3

UNITS: FORCES IN KIPS
DISPLACEMENTS IN FEET
NOTES: + TENSION
- COMPRESSION

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
DISPLACEMENTS AND FORCES

FIGURE 3H-21



LOSS OF FAN – STEADY STATE

U = D + L' + A + T4

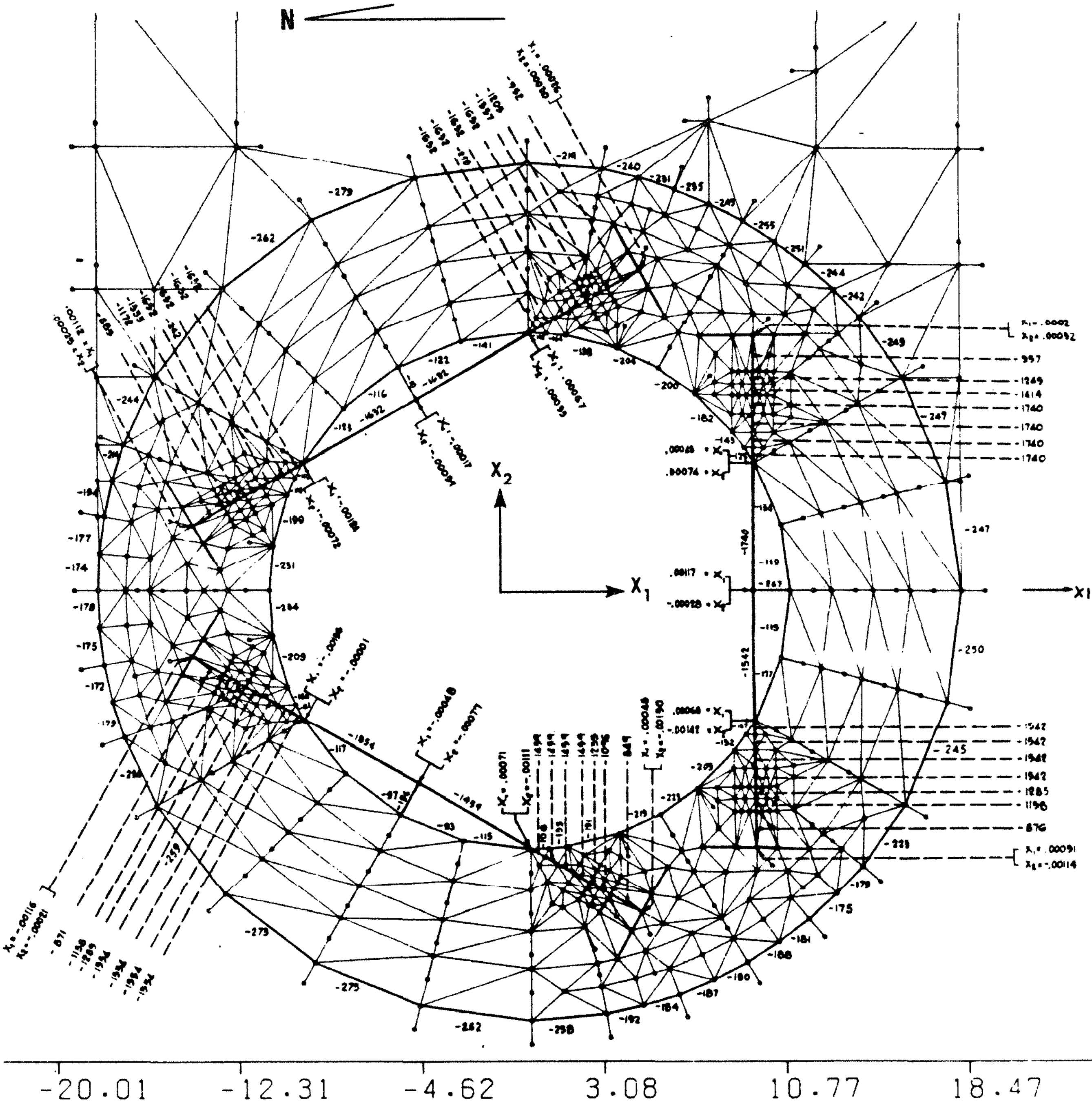
UNITS: FORCES IN KIPS
DISPLACEMENTS IN FEET

NOTES: + TENSION
- COMPRESSION

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
DISPLACEMENT AND FORCES – SHEET 3

FIGURE 3H-22



START-UP TRANSIENT

U = D + L' + A + T2

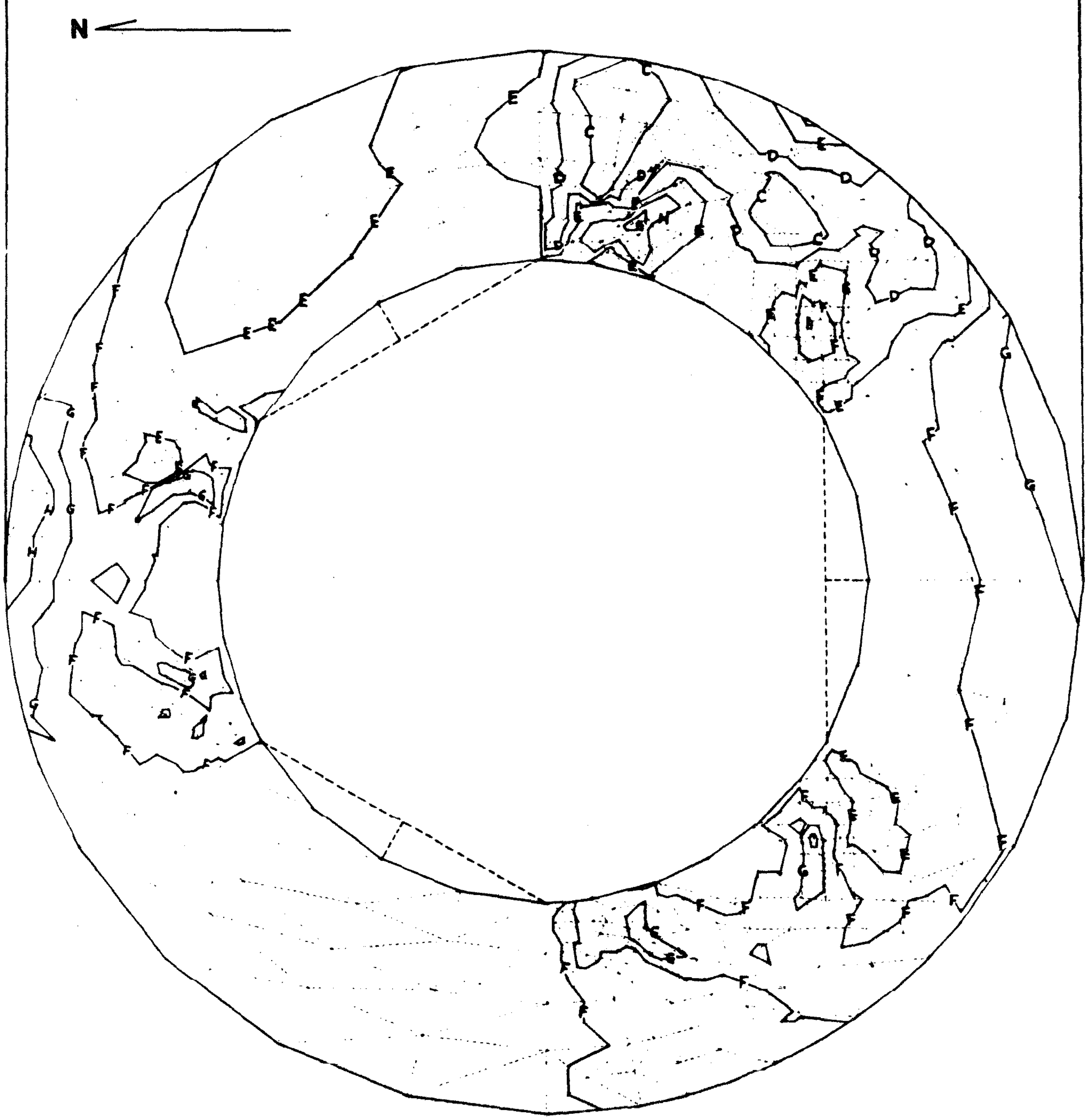
UNITS: FORCES IN KIPS
DISPLACEMENTS IN FEET

NOTES: + TENSION
− COMPRESSION

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
DISPLACEMENT AND FORCES – SHEET 2

FIGURE 3H-23



NORMAL OPERATING – STEADY STATE

$\overline{U} = 1.5\ (D + L' + A + T1)$

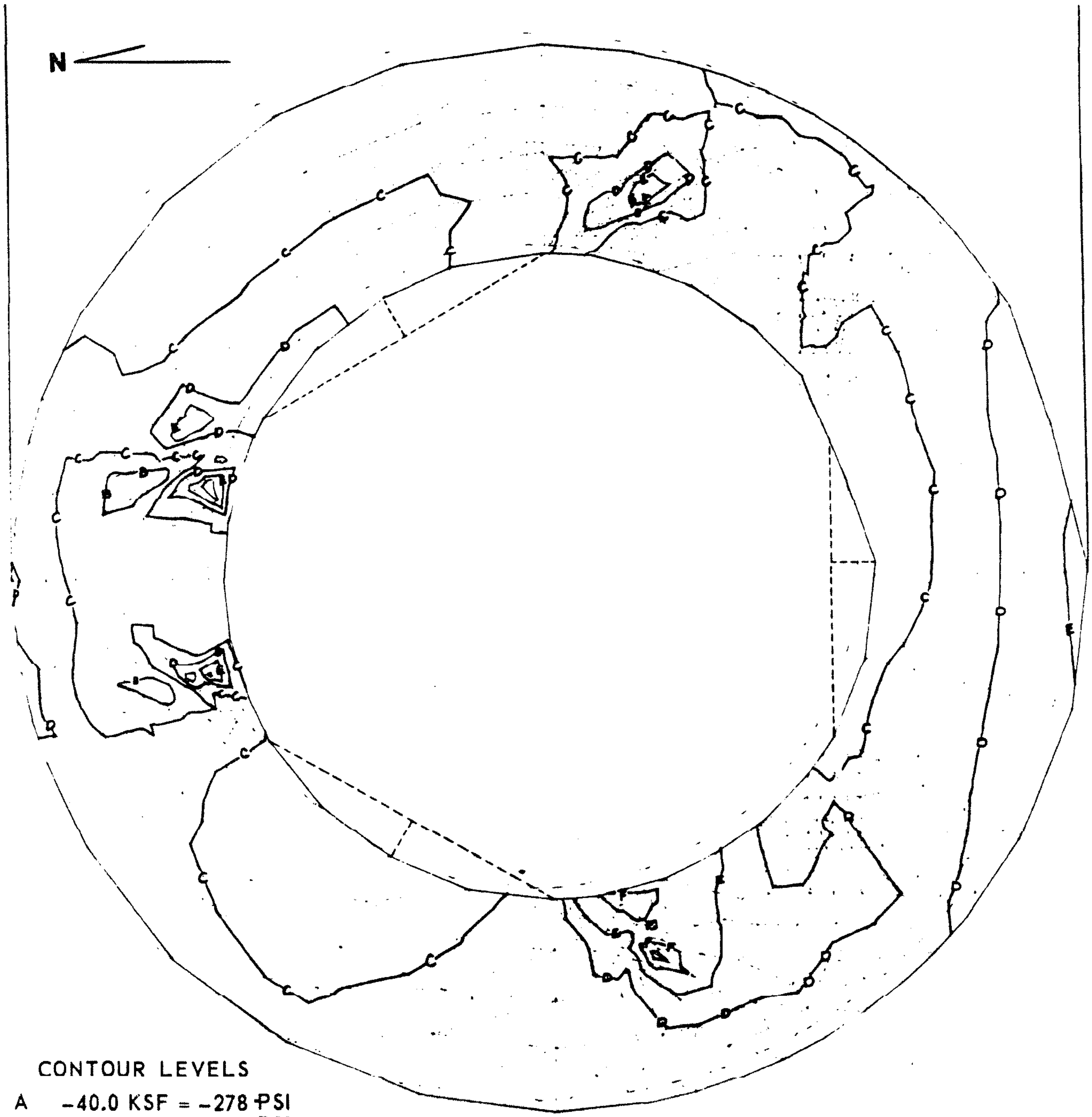
CONTOUR LEVELS

A	−50.0 KSF =	−347 PSI
B	−40.0 KSF =	−278 PSI
C	−30.0 KSF =	−208 PSI
D	−20.0 KSF =	−139 PSI
E	−10.0 KSF =	−69 PSI
F	0.0	
G	10.0 KSF =	69 PSI
H	20.0 KSF =	139 PSI
I	30.0 KSF =	208 PSI
J	40.0 KSF =	278 PSI
K	50.0 KSF =	347 PSI

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
PRINCIPAL STRESS PLOT – SHEET 1

FIGURE 3H-24



CONTOUR LEVELS

A	−40.0 KSF = −278 PSI	
B	−20.0 KSF = −139 PSI	
C	0.0	
D	20.0 KSF =	139 PSI
E	40.0 KSF =	278 PSI
F	60.0 KSF =	417 PSI
G	80.0 KSF =	556 PSI
H	100. KSF =	694 PSI
I	120. KSF =	833 PSI
J	140. KSF =	972 PSI

LOCA – STEADY STATE

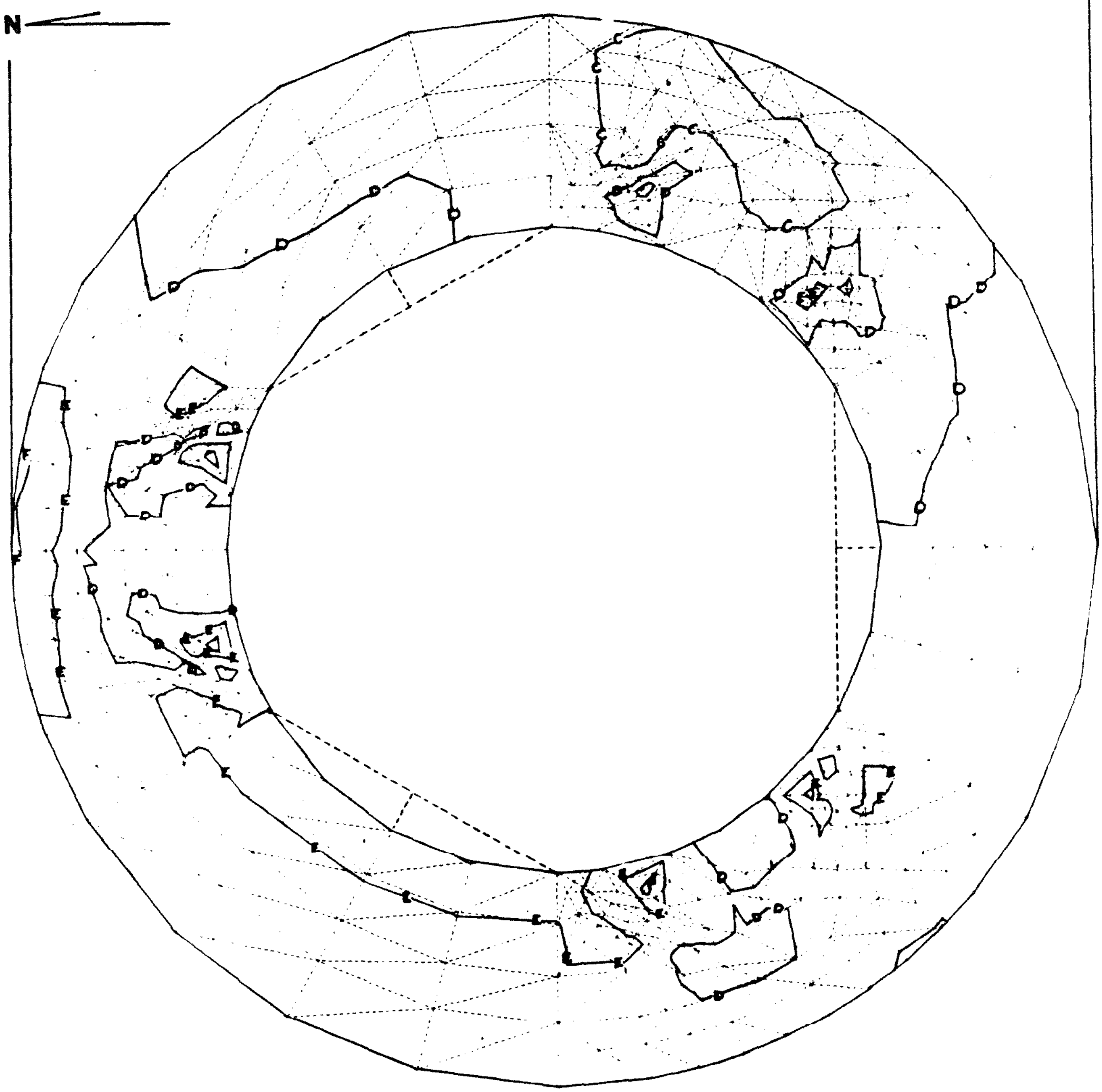
U = T1 + 1.15 (D + L′ + A + P + Q + DBE)

LOCA – SOUTH HOT LEG GUILLOTINE BREAK
(CRITICAL LOCA LOADING)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
PRINCIPAL STRESS PLOT – SHEET 2

FIGURE 3H-25



LOSS OF FAN – STEADY STATE
U = D + L' + A + T4

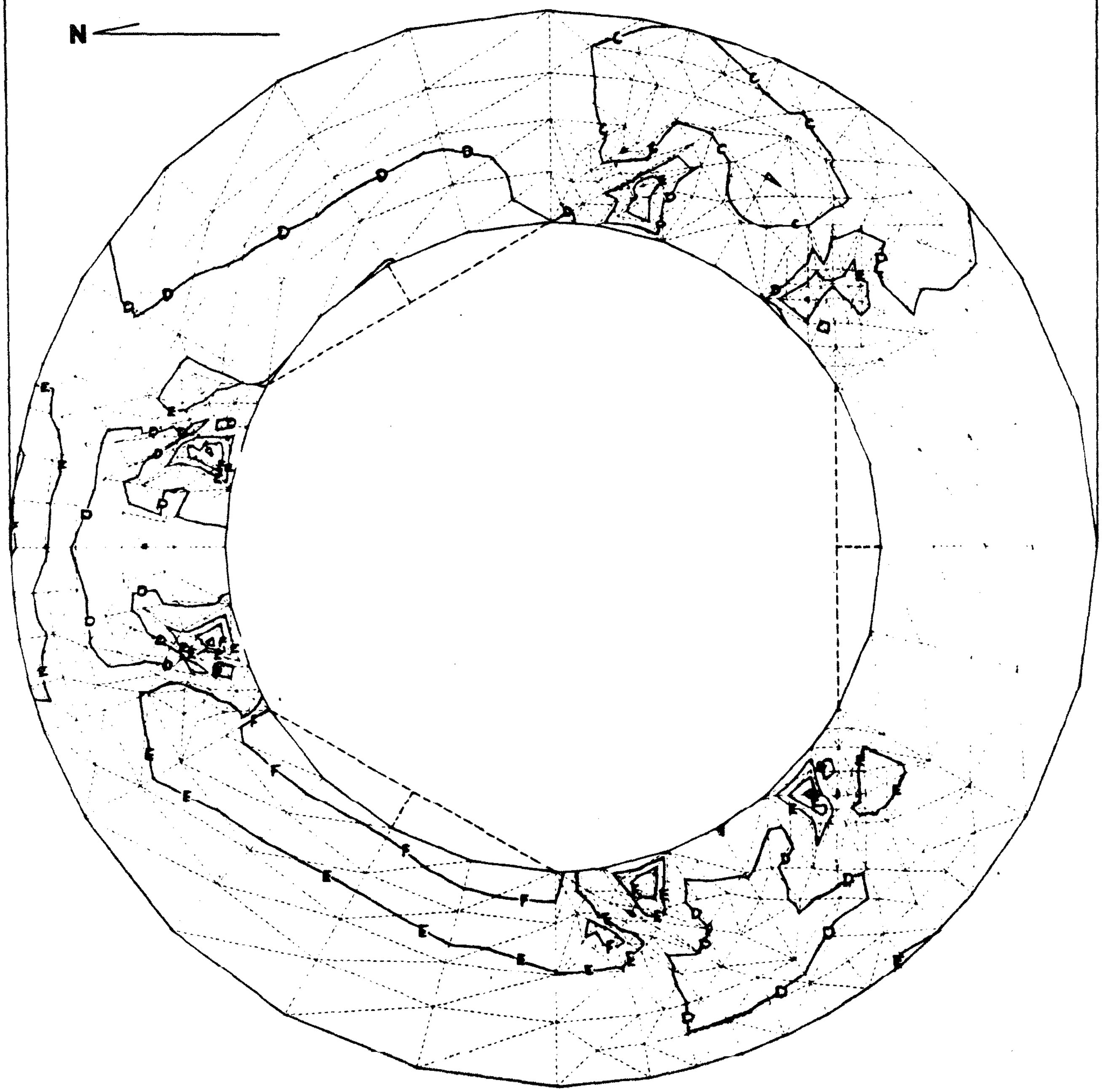
CONTOUR LEVELS

A	−60.0 KSF =	−417 PSI
B	−40.0 KSF =	−278 PSI
C	−20.0 KSF =	−139 PSI
D	0.0	
E	20.0 KSF =	139 PSI
F	40.0 KSF =	278 PSI
G	60.0 KSF =	417 PSI
H	80.0 KSF =	556 PSI.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
PRINCIPAL STRESS PLOT – SHEET 3

FIGURE 3H-26



POST LOCA - STEADY STATE

$U = D + L' + A + T3$

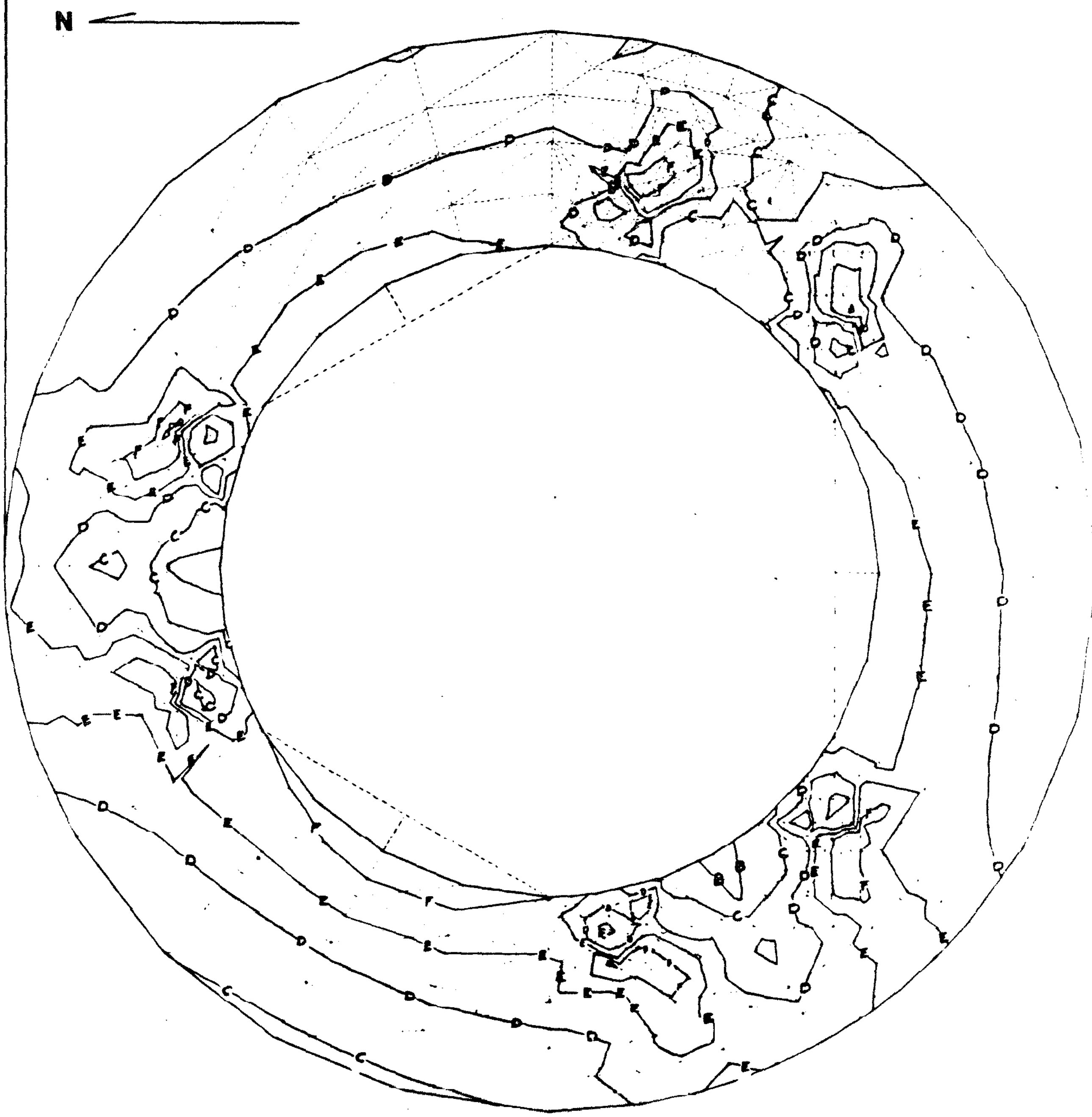
CONTOUR LEVELS

A	−60.0 KSF	=	−417 PSI
B	−40.0 KSF	=	−278 PSI
C	−20.0 KSF	=	−139 PSI
D	0.0		
E	20.0 KSF	=	139 PSI
F	40.0 KSF	=	278 PSI
G	60.0 KSF	=	417 PSI
H	80.0 KSF	=	556 PSI
I	100. KSF	-	694 PSI
J	120. KSF	=	833 PSI

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
PRINCIPAL STRESS PLOT – SHEET 4

FIGURE 3H-27



CONTOUR LEVELS

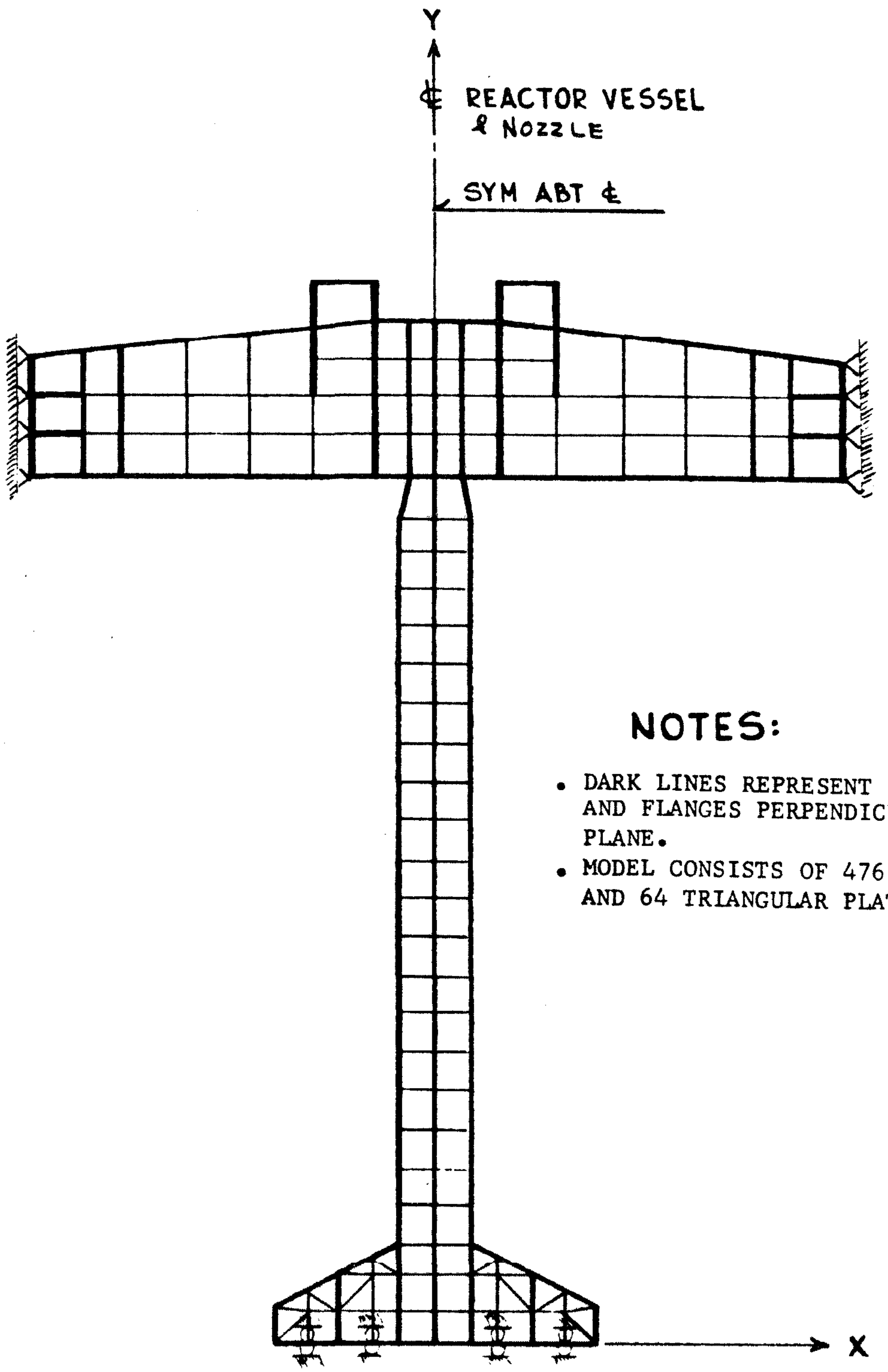
A	−10.0 KSF = 69 PSI
B	0.0
C	10.0 KSF = 69 PSI
D	20.0 KSF = 139 PSI
E	30.0 KSF = 208 PSI
F	40.0 KSF = 278 PSI
G	50.0 KSF = 347 PSI
H	60.0 KSF = 417 PSI
I	70.0 KSF = 486 PSI
J	80.0 KSF = 556 PSI

START-UP TRANSIENT $U = D + L' + A + T2$

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

REACTOR CAVITY WALL
PRINCIPAL STRESS PLOT – SHEET 5

FIGURE 3H-28



NOTES:

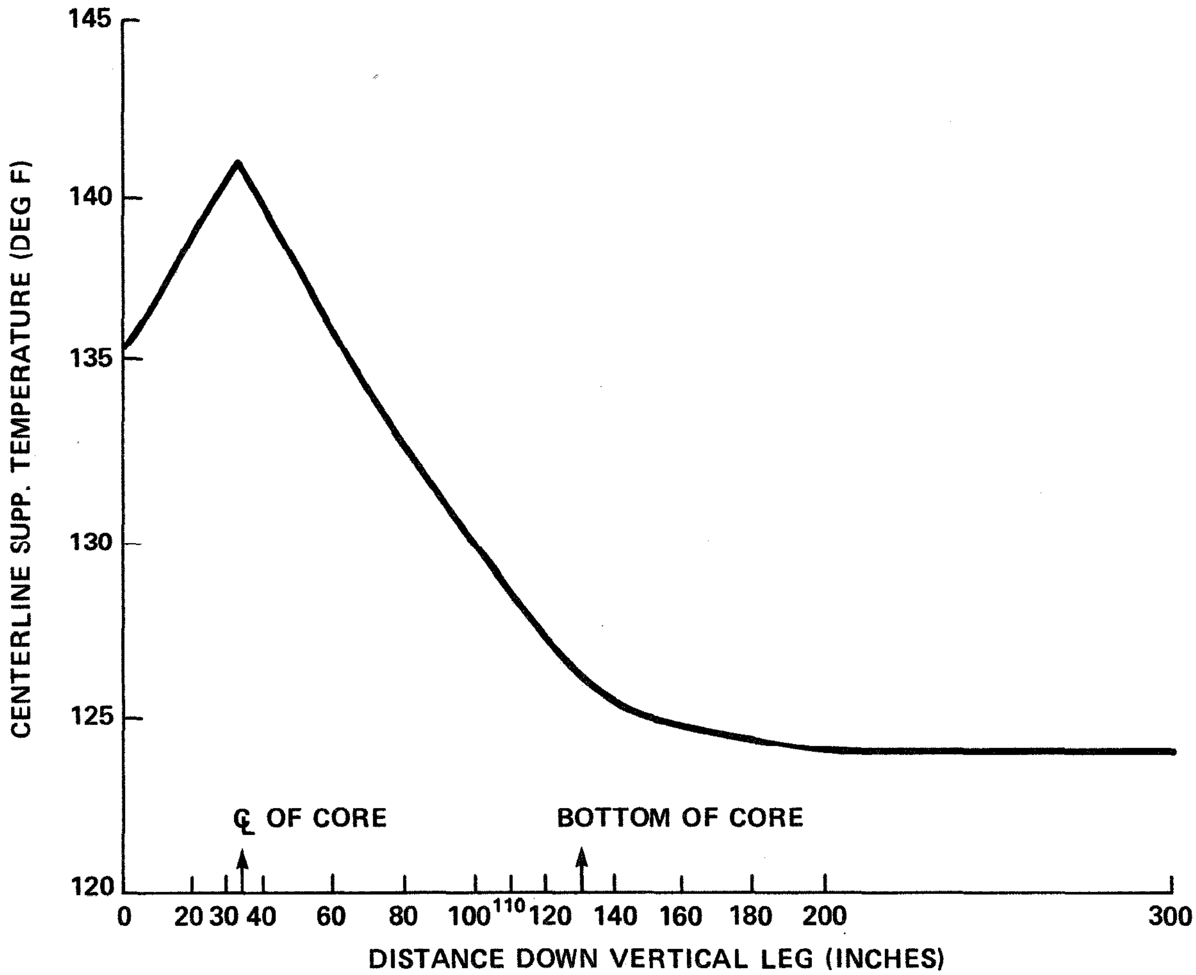
- DARK LINES REPRESENT STIFFENERS AND FLANGES PERPENDICULAR TO XY PLANE.
- MODEL CONSISTS OF 476 QUADRILATERAL AND 64 TRIANGULAR PLATE ELEMENTS.

ELEVATION

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT

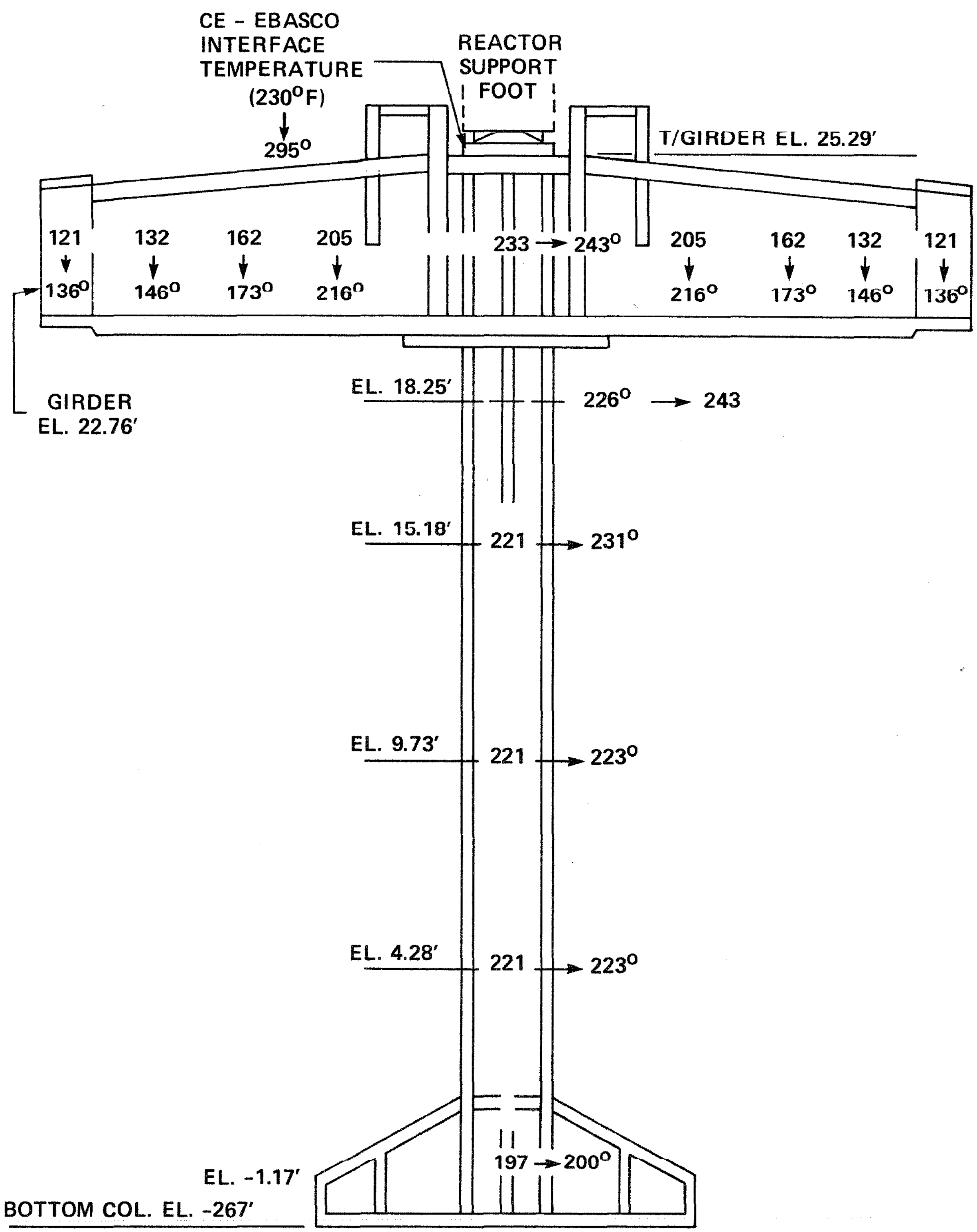
BEAM – COLUMN REACTOR SUPPORT FINITE ELEMENT MODEL

FIGURE 3H-29



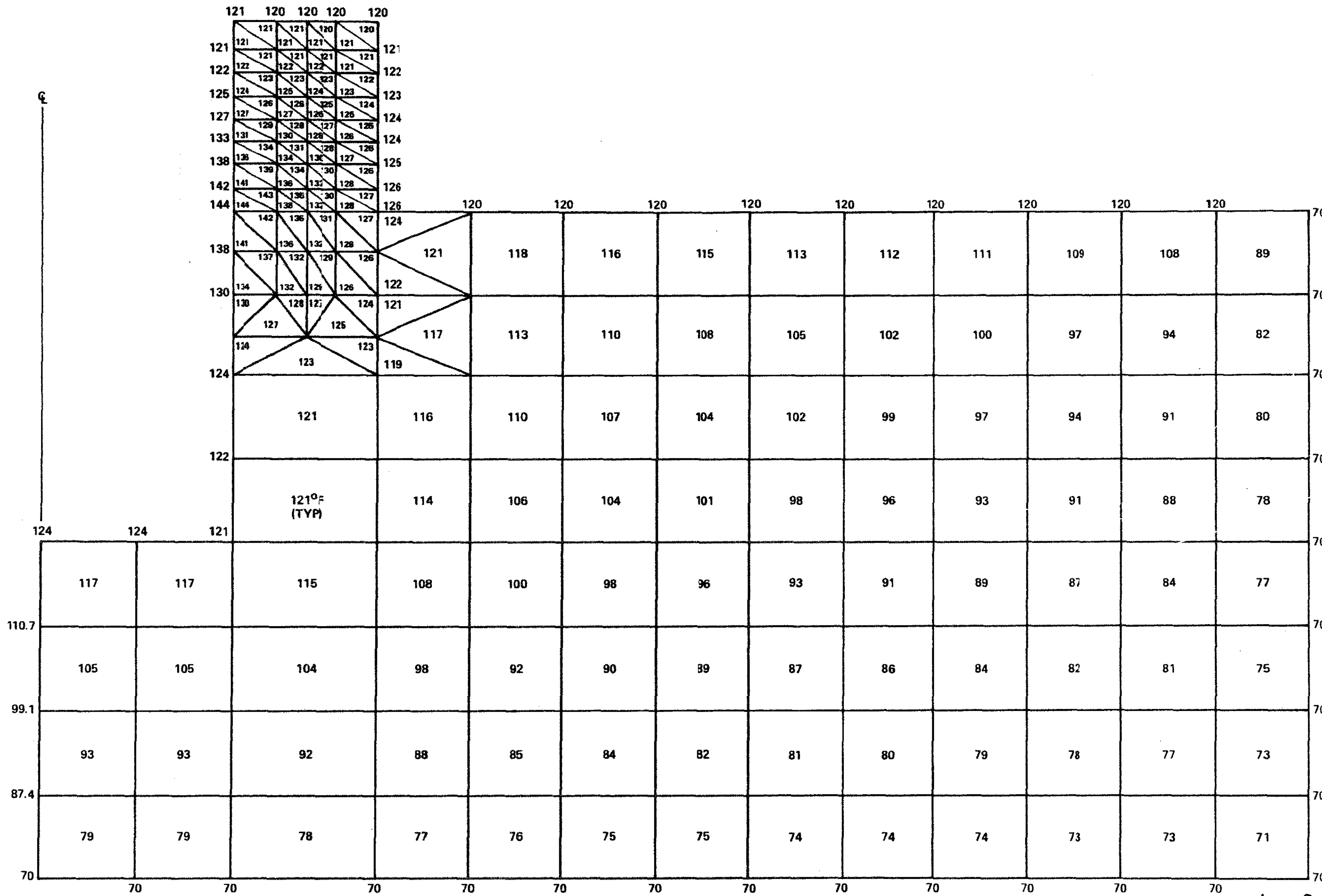
Am. 3-7/85

FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	VERTICAL RV SUPPORT LEG TEMPERATURES	Figure 3H-30



Am. 3-7/85

FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	REACTOR SUPPORT TEMPERATURE DISTRIBUTION LONG TERM LOSS OF FANS TIME = 4.5 HOURS AIR TEMPERATURE = 295°F (45 MIN)	Figure 3H-31

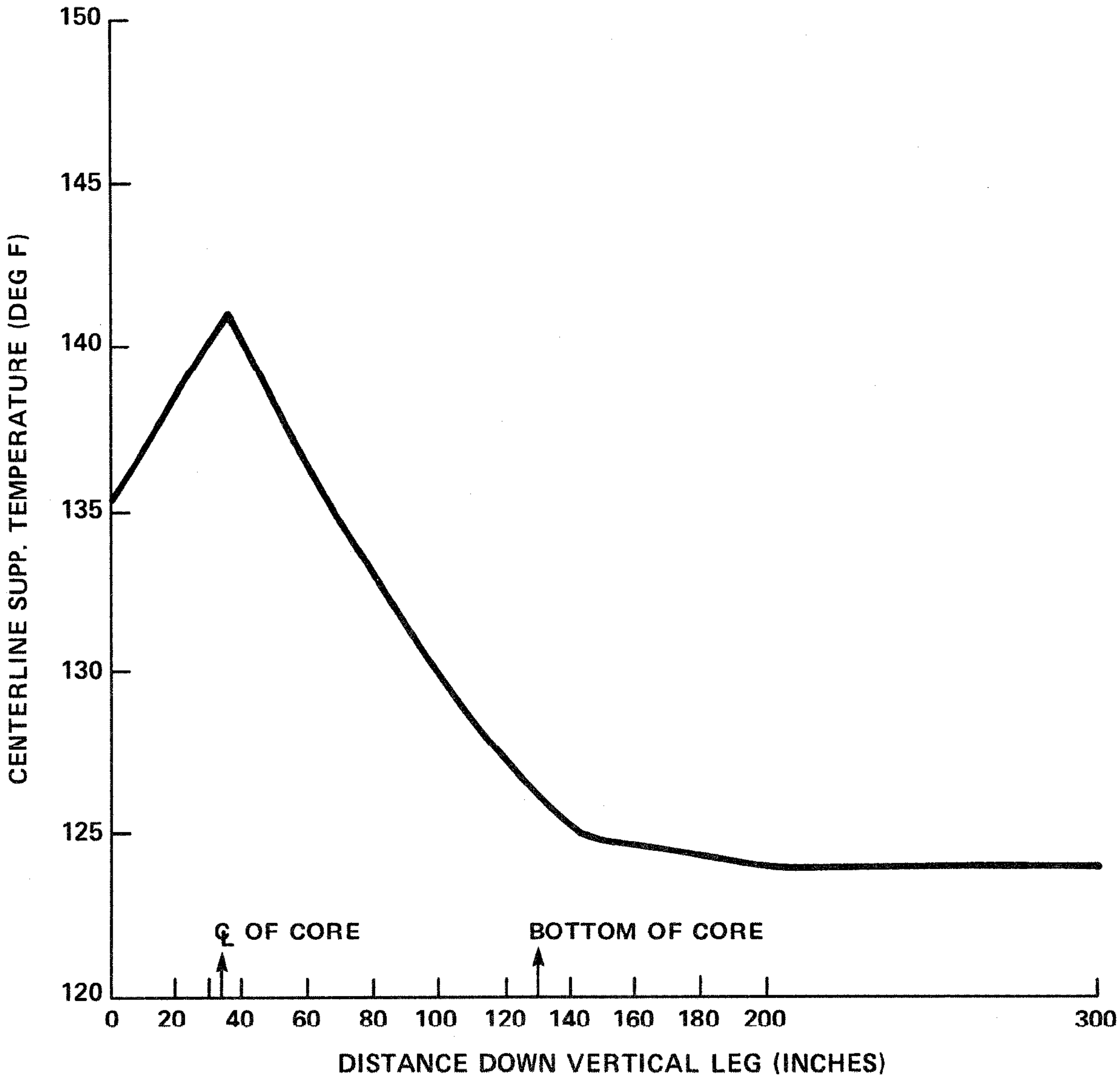


Am. 3-7/85

VERTICAL FINITE ELEMENT MODEL WITH
NORMAL TEMPERATURE DISTRIBUTION

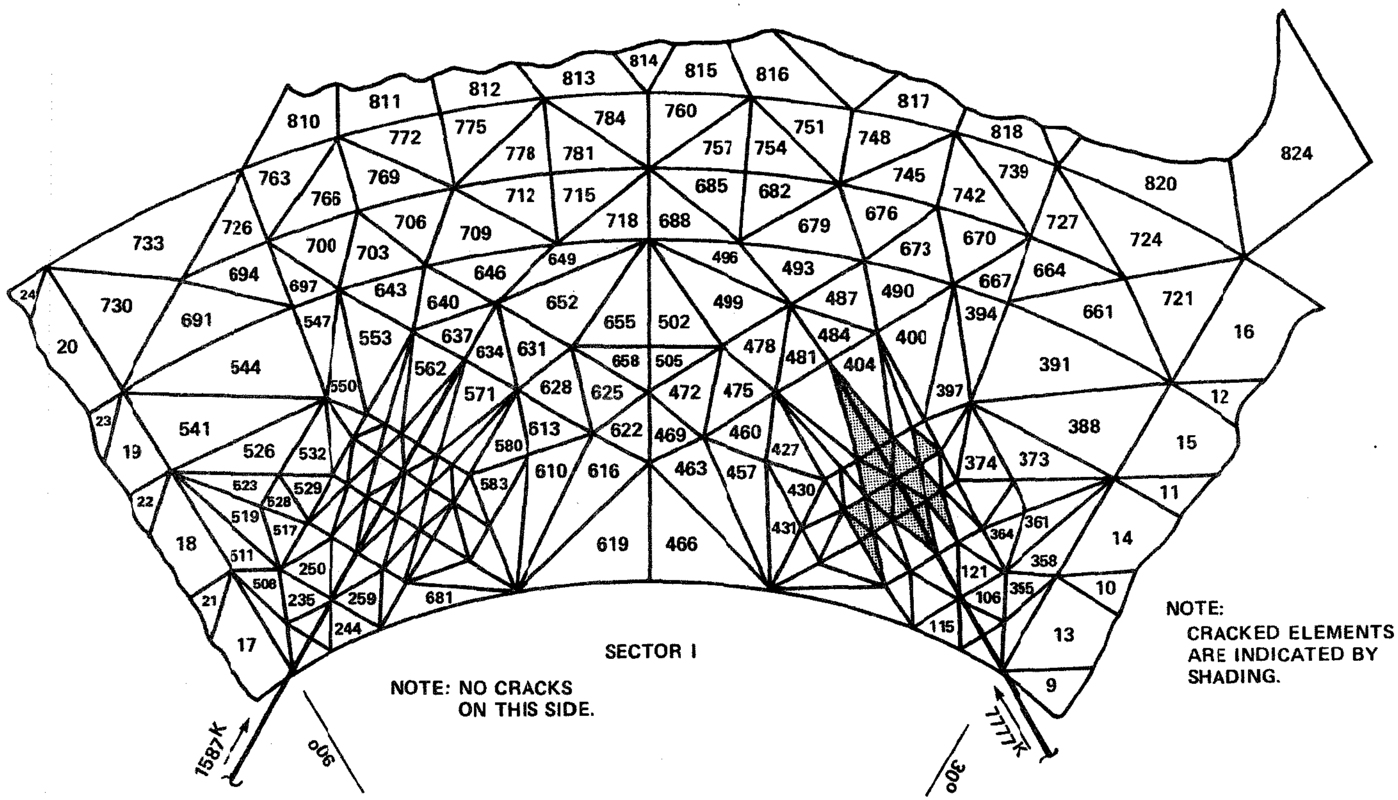
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

VERTICAL FINITE ELEMENT MODEL
FIGURE 3H-32



Am. 3-7/85

FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	VERTICAL RV SUPPORT LEG TEMPERATURES	Figure 3H-33

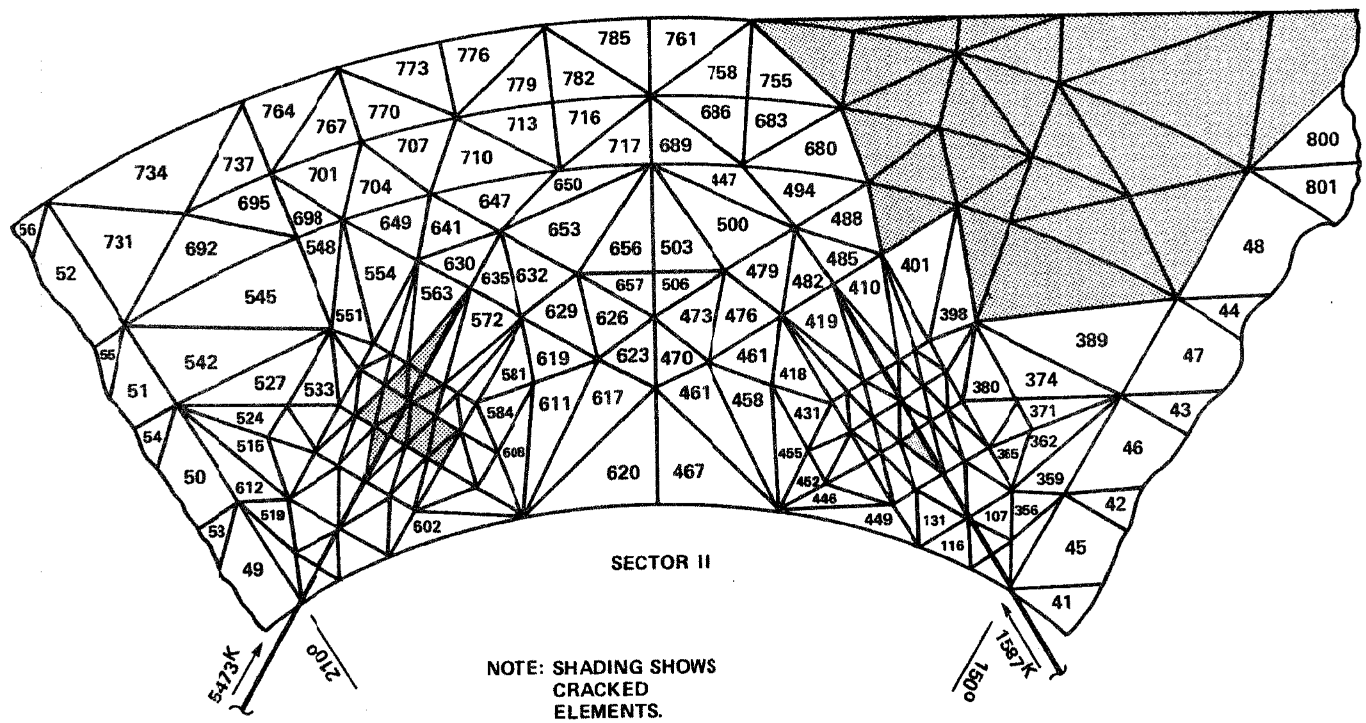


Am. 3-7/85

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

HORIZONTAL MODEL CRACKING PATTERN
FOR SECTOR I

FIGURE 3H-34



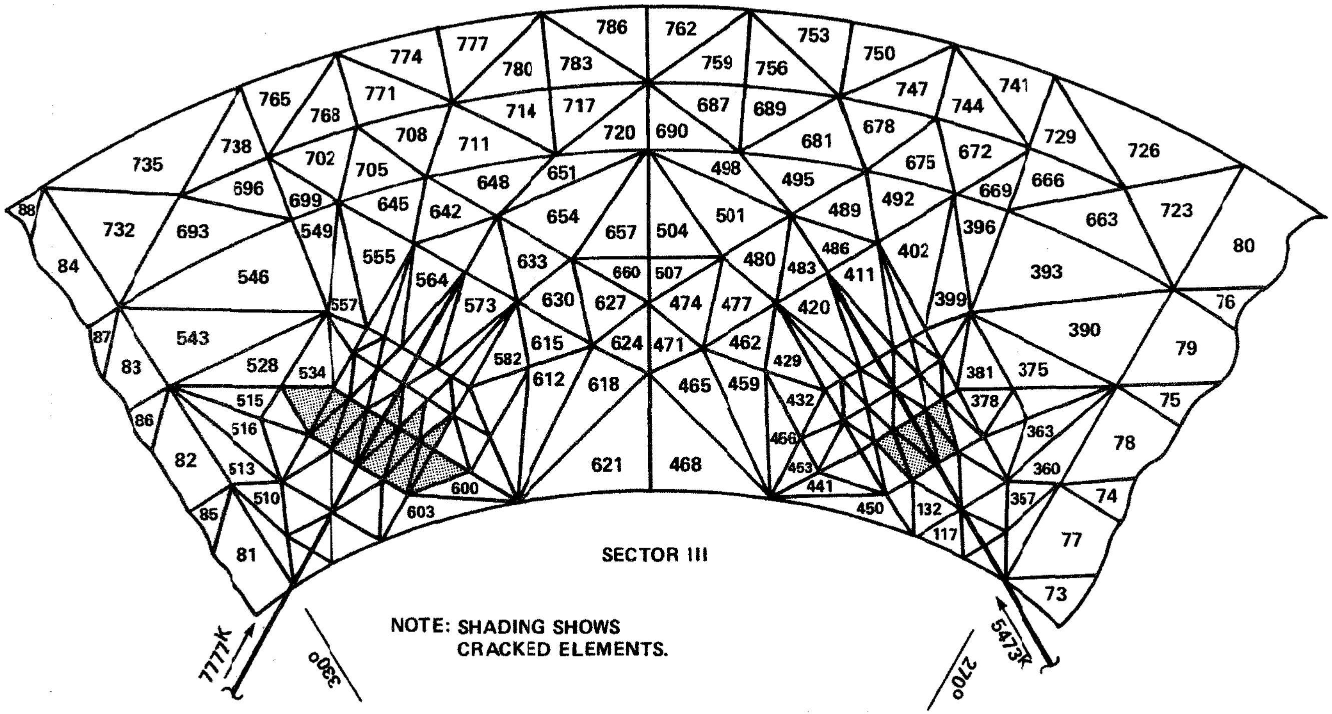
NOTE: SHADING SHOWS CRACKED ELEMENTS.

Am. 3-7/85

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

HORIZONTAL MODEL CRACKING PATTERN FOR SECTOR II

FIGURE 3H-35



Am. 3-7/85

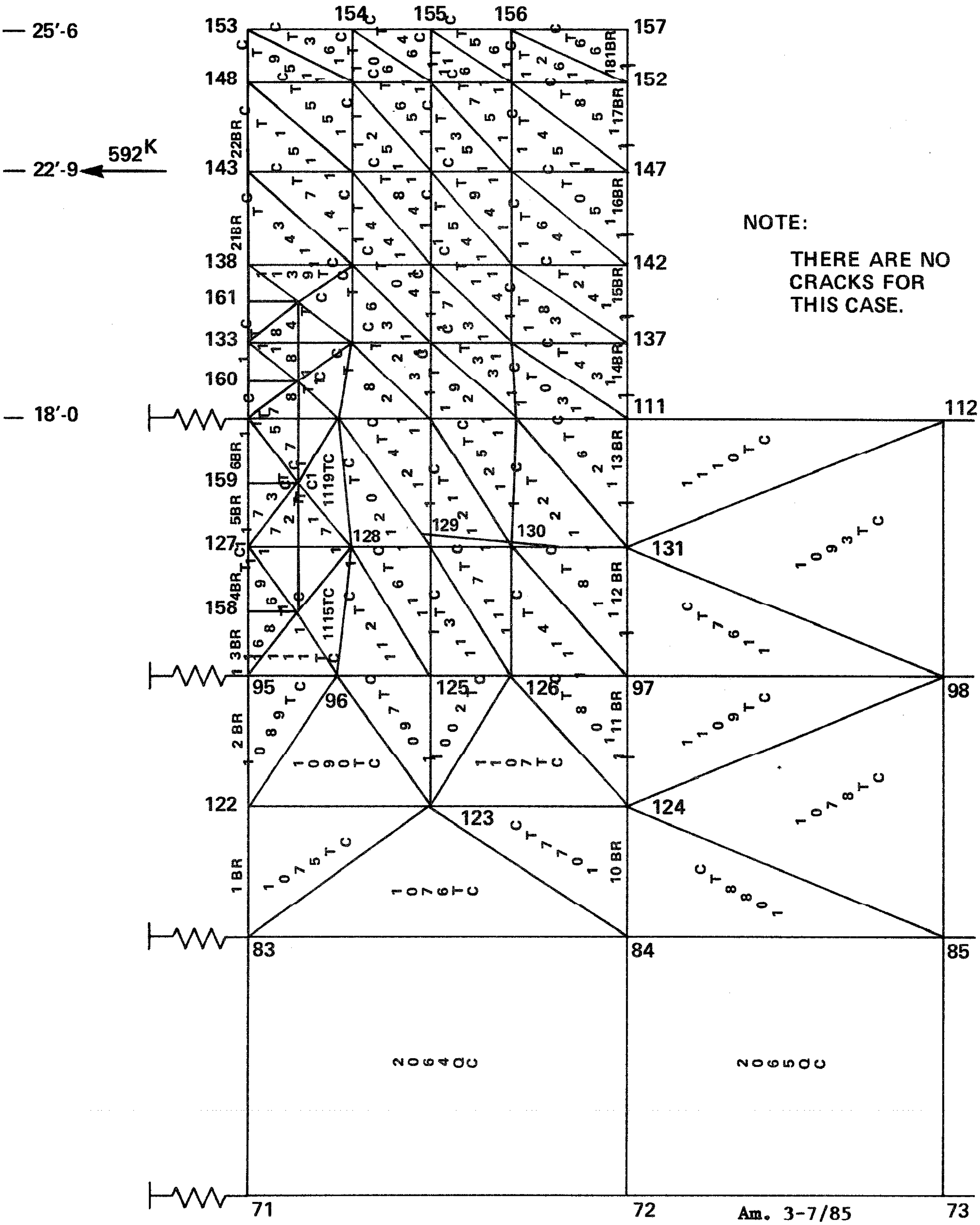
FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

HORIZONTAL MODEL CRACKING PATTERN FOR SECTOR III

FIGURE 3H-36

TOTAL LOAD = -592^{K} (SECTION 1)

(REF. - COMPUTER RUN# VKGAJDS, DT. 3/16/84)



NOTE:

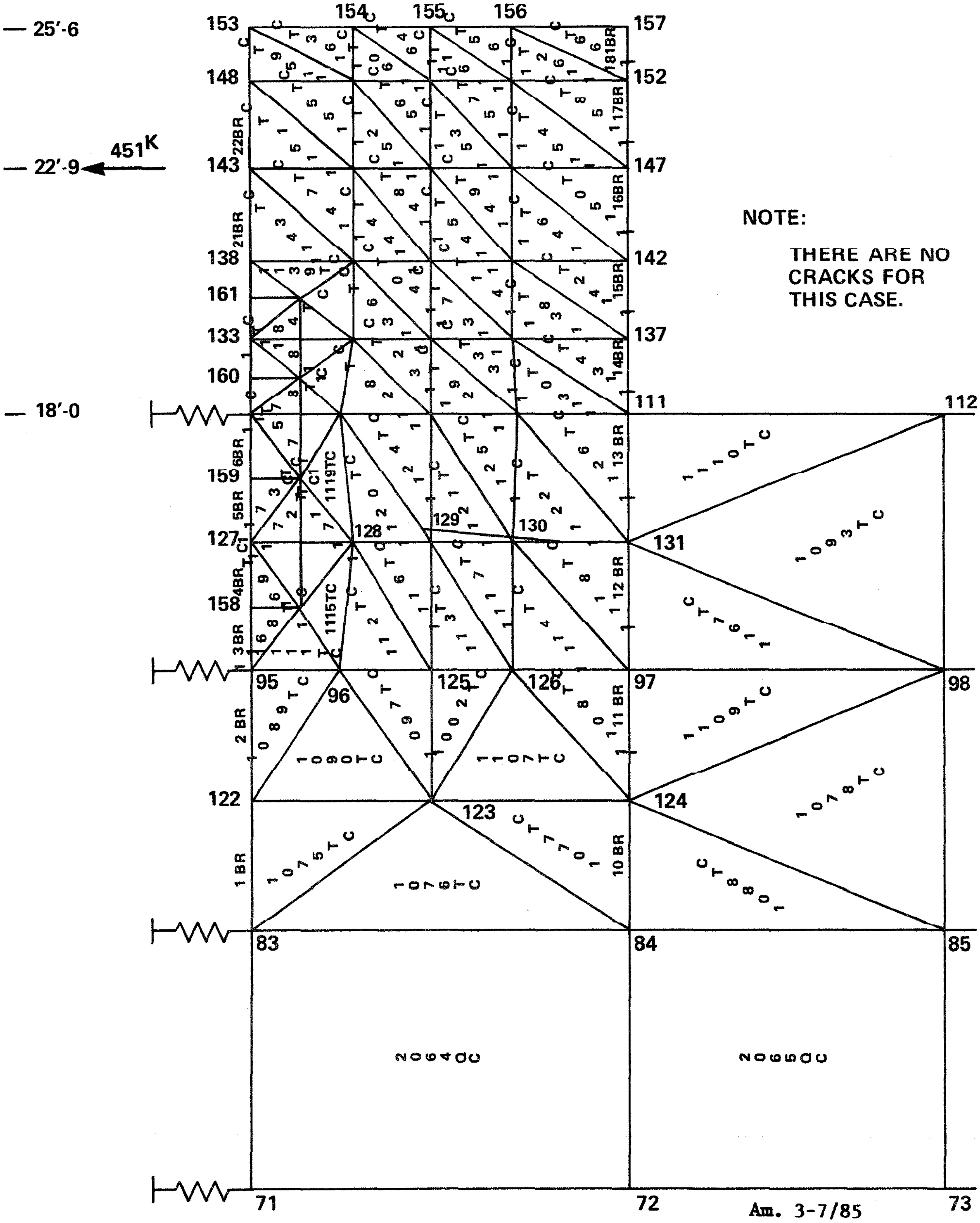
THERE ARE NO CRACKS FOR THIS CASE.

Am. 3-7/85

FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LOAD SUBCASE 1	Figure 3H-37

TOTAL LOAD = -451^K (SECTOR 1-2)

(REF. - COMPUTER RUN# VKGAJDS, DT. 3/16/84)

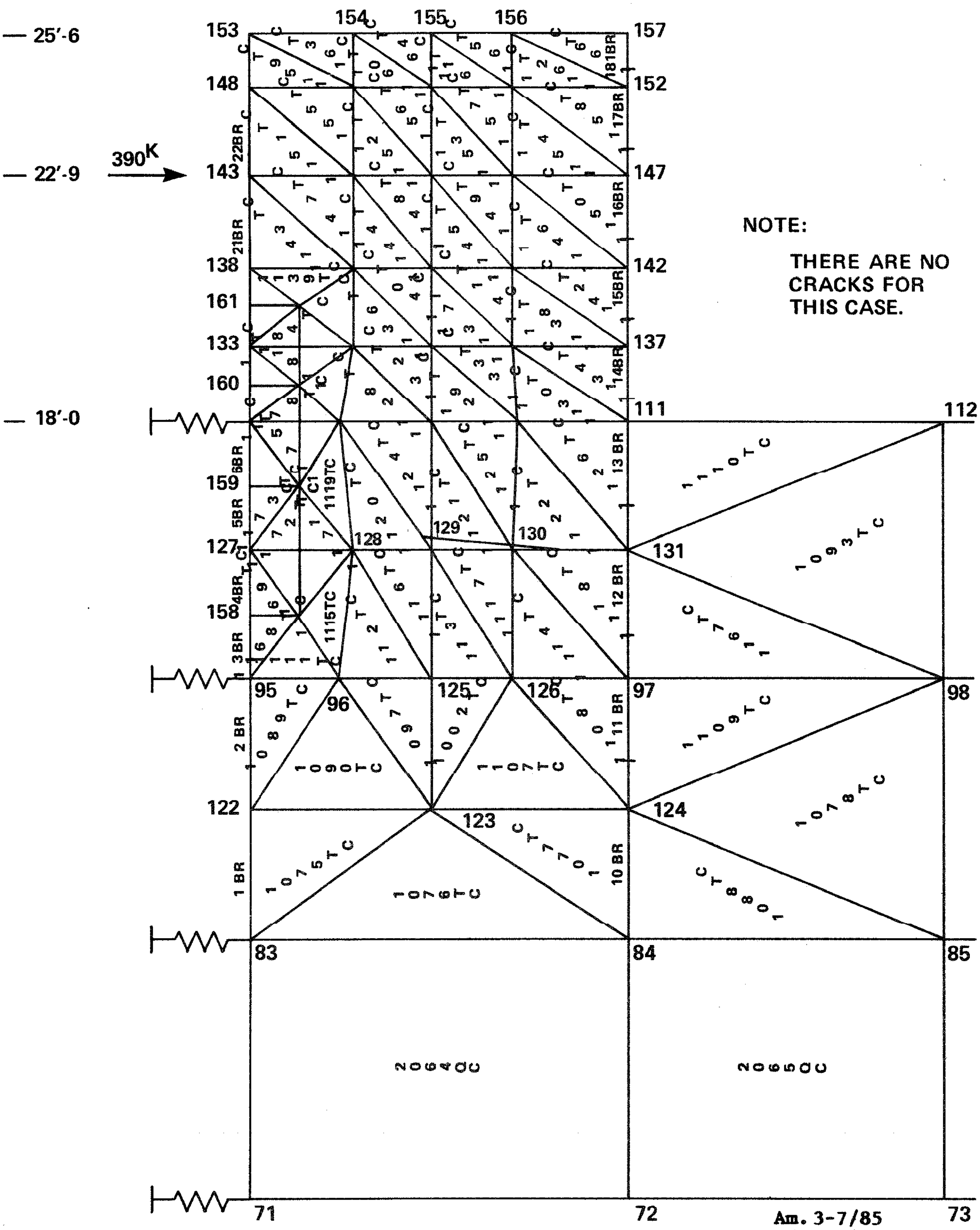


Am. 3-7/85

FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LOAD SUBCASE 4	Figure 3H-38

TOTAL LOAD = 390^K (SECTOR 2)

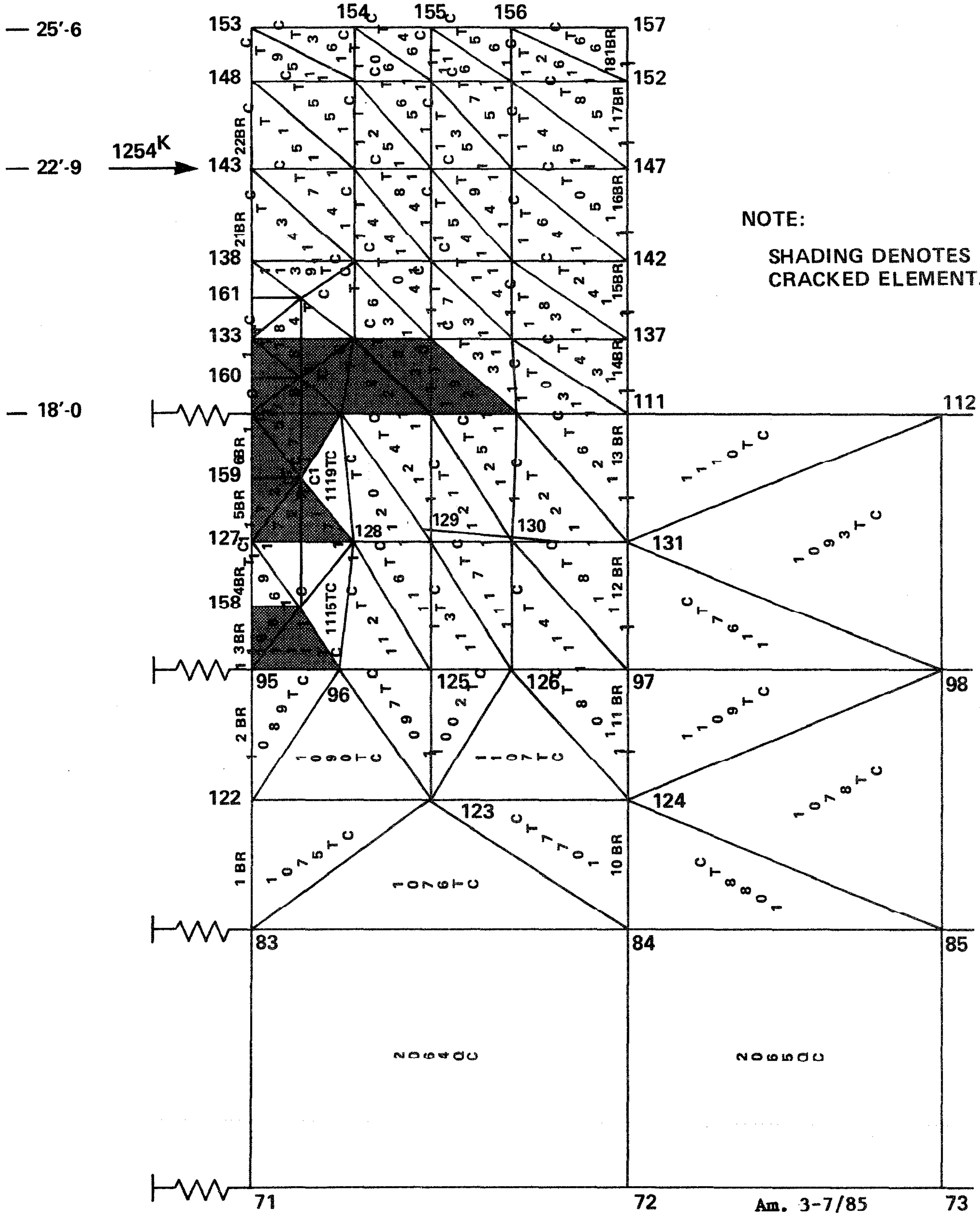
(REF. - COMPUTER RUN# VKGAJDS, DT. 3/16/84)



Am. 3-7/85

FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LOAD SUBCASE 2	Figure 3H-39

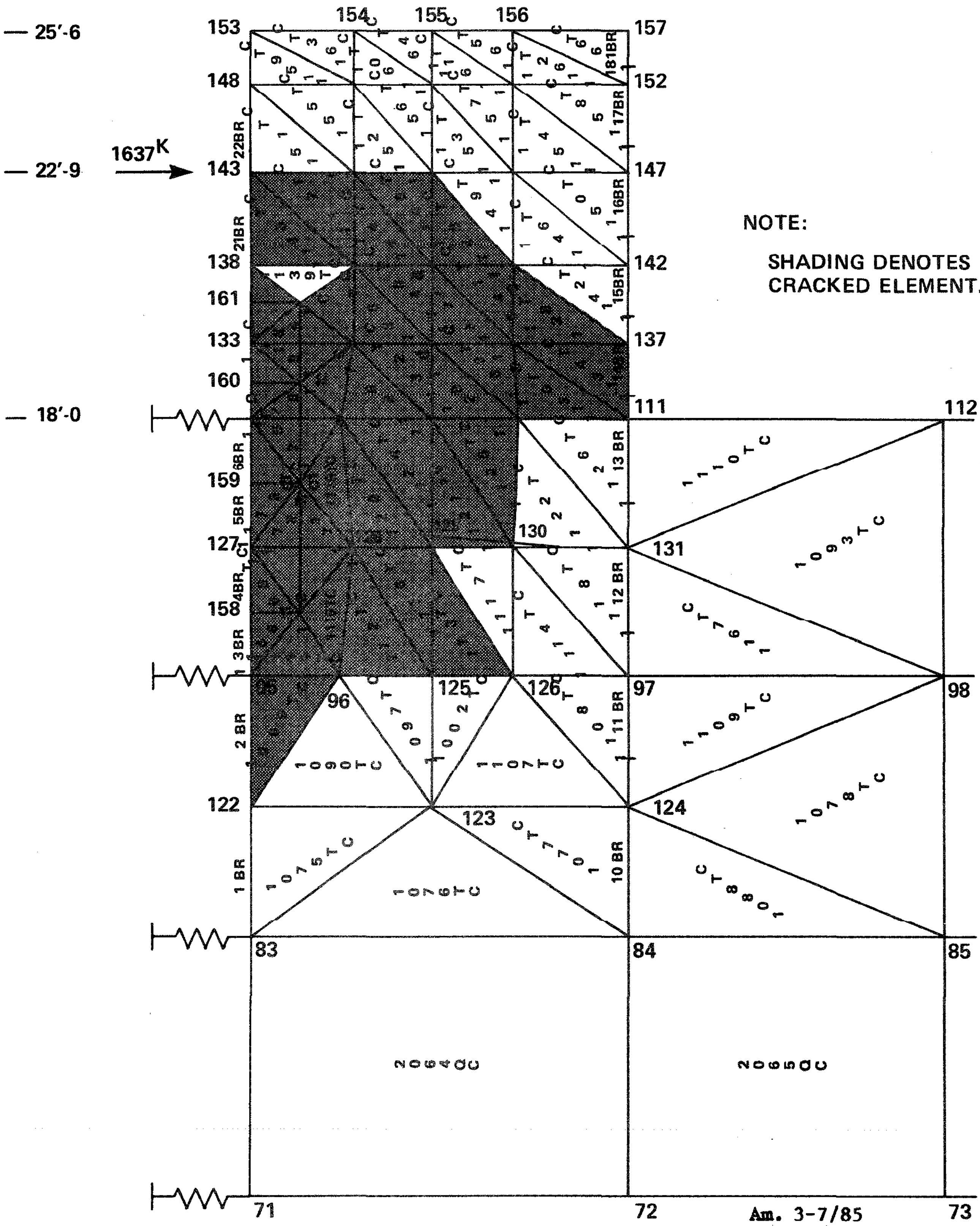
TOTAL LOAD = 1254^K (SECTOR 2-3)
(REF. - COMPUTER RUN# VKGAJDS, DT. 3/16/84)



FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LOAD SUBCASE 5	Figure 3H-40

TOTAL LOAD = 1637^{K} (SECTOR 3))

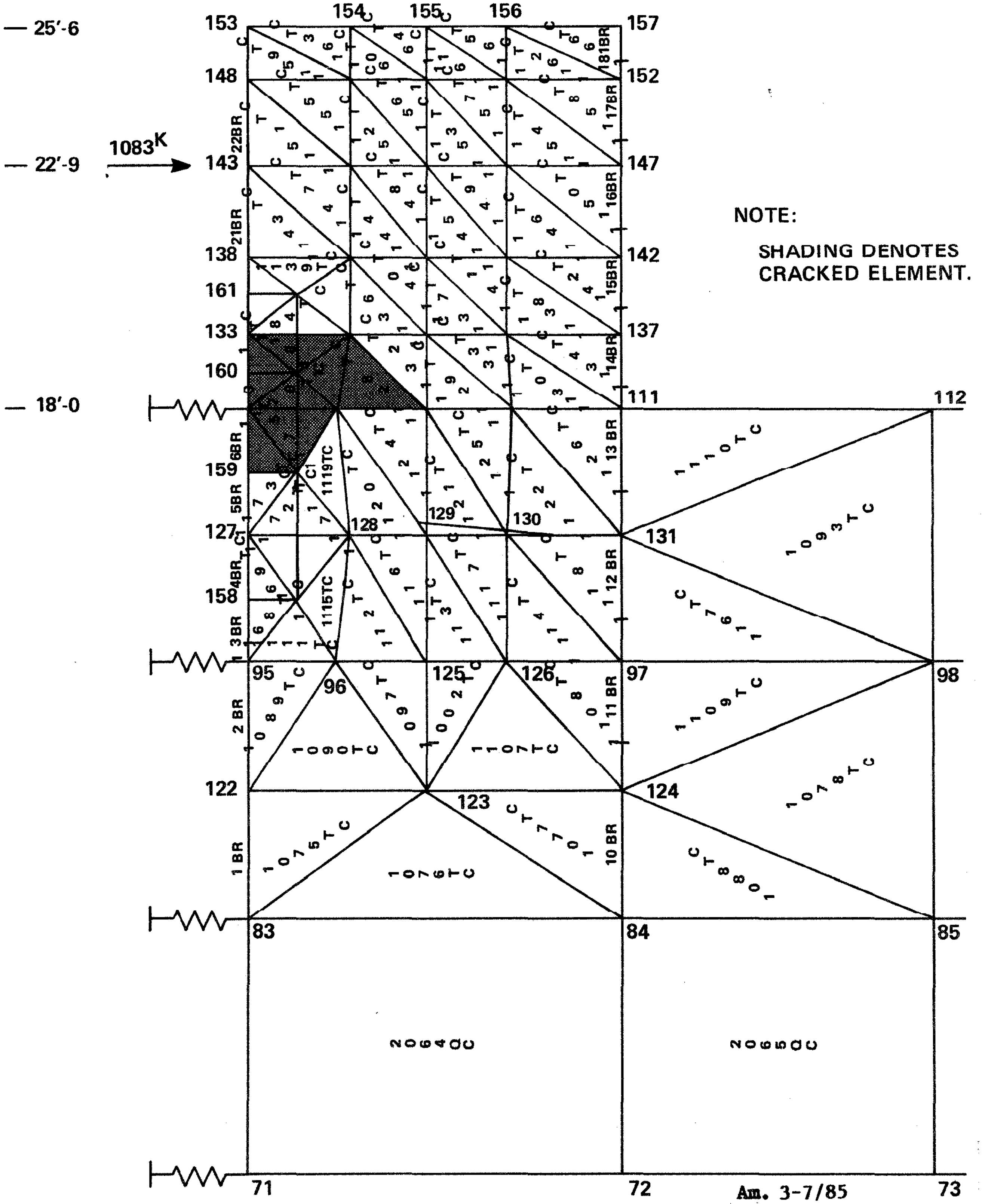
(REF. - COMPUTER RUN# VKGAJDS, DT. 3/16/84)



FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LOAD SUBCASE 3	Figure 3H-41

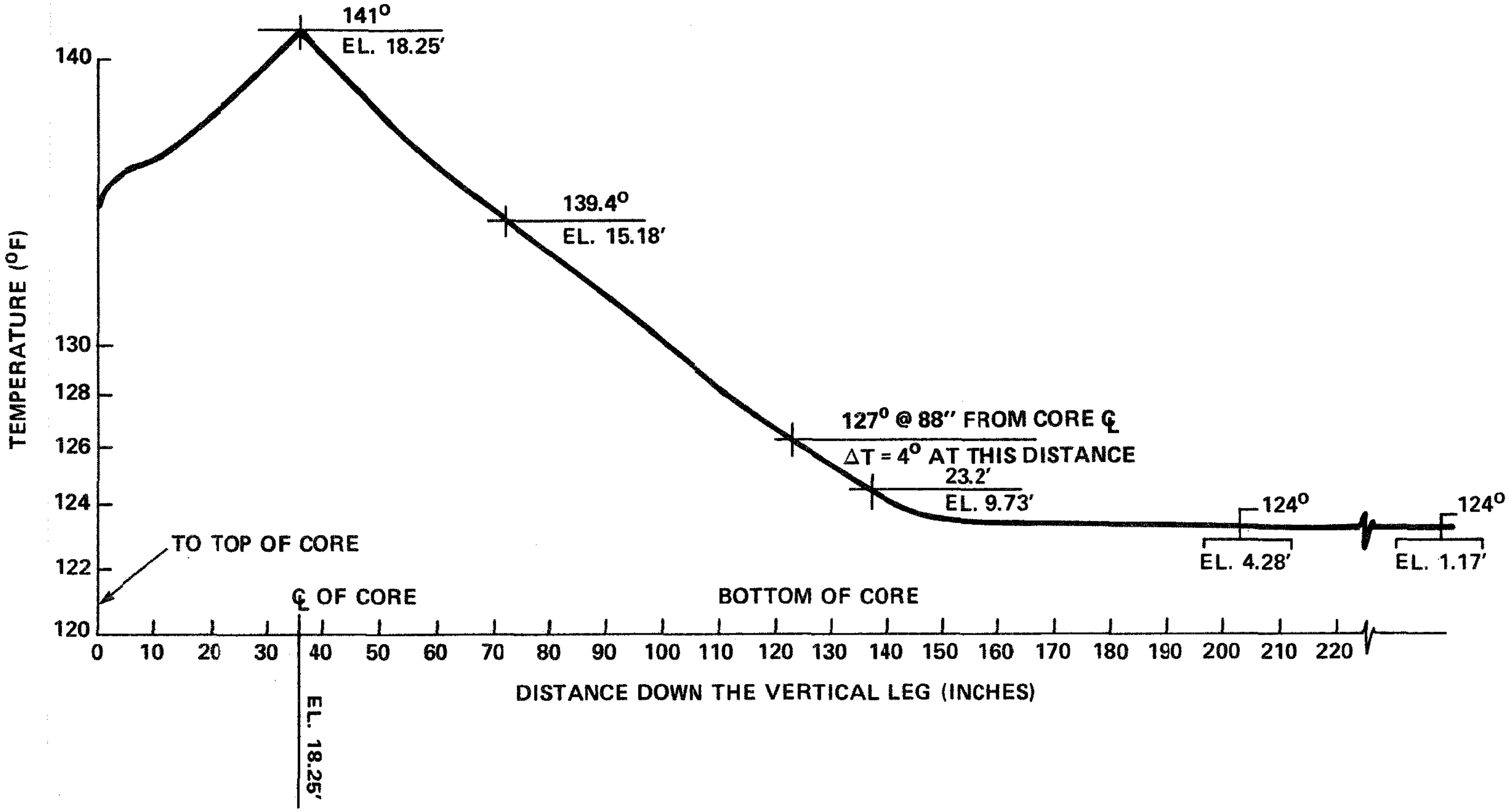
TOTAL LOAD = 1083^K (SECTOR 3-1)

(REF. - COMPUTER RUN# VKGAJDS, DT. 3/16/84)



Am. 3-7/85

FLORIDA POWER & LIGHT CO. St. Lucie Plant Unit 1	LOAD SUBCASE 6	Figure 3H-42



℄ CORE @ EL. 18.25'

℄ SUPP "A" @ EL. 22.76' – AVG TEMP = 135°F
GIRD

Am. 3-7/85

FLORIDA POWER & LIGHT CO.
St. Lucie Plant
Unit 1

LONG TERM STEADY STATE NORMAL OPERATION

Figure 3H-43

ATTACHMENT 1

TO

APPENDIX 3H

ATTACHMENT 1

Thermal Analyses

A. INTRODUCTION

The temperature distributions used in the reactor support stress analyses have been generated by using the three-dimensional thermal analysis code LION. This program utilizes a "thermal circuit" approach to analyze both steady state and transient problems. With this technique the heat conducting body is broken up into a complex of "thermal" resistors, the impedances being defined by the geometry and the thermal conductivity of the material. Similarly, the thermal capacitance is defined as the product of the specific heat, density and volume of the material. Thus, a set of matrix equations is defined which is analogous to those used in electrical circuit problems where temperature is now used instead of voltage.

Heat transfer coefficients and boundary conditions (temperatures) were defined in a manner which is conservative for this analysis. Refer to Tables 2 and 3 for these parameter values.

B. THERMAL MODELS

Thermal models have been constructed for both the reactor steel beam-column assembly and the concrete primary shield wall. Both models overlap in the regions where the steel support structures are embedded in the shield wall allowing realistic interfacing. The assumed properties of the concrete and steel are listed in Table 1.

The shield wall model consists of a 183 element, two-dimensional, annular concrete slab at elevation +22.5 feet where the three horizontal members of the reactor support structure are embedded. The shield wall model includes the horizontal steel members. The assumed boundary conditions are:

1. Thermal insulation in the vertical direction (vertical temperature gradients within the concrete are ignored)

2. Boundary conditions at the inner surface of the shield wall (usually radiant heat transfer, convection, etc.)

3. Boundary conditions at the outer surface of the shield wall (usually natural convection)

4. Interface temperature at the midpoint of the horizontal member of the reactor support (as obtained from the reactor beam-column model).

The reactor beam-column model consists of 271 elements which include concrete interface regions at the floor of the reactor cavity the region of the primary shield wall near the 22.5 foot elevation, and the reactor support foot up to the reactor vessel nozzle. The imposed boundary conditions are:

1. Concrete temperature three feet below the floor of the cavity.

2. Concrete temperature within the center of the shield wall.

3. Reactor coolant temperature at the inner surface of the reactor nozzle.

4. Reactor support steel surface boundary conditions (usually natural convection and radiant heat transfer).

In addition to the reactor support structure temperature distribution for stress analysis purposes, reactor coolant system/support structure interface temperature is obtained from this model. This temperature is defined at the bottom of the lubrication plate.

C. <u>NORMAL OPERATION AND STARTUP CASES</u>

Temperature distributions have been obtained for both the long term operation and cold startup conditions. These cases are used as initial conditions for the accident cases described in the next section. Both conditions are characterized by the operation of the reactor cavity and reactor support cooling systems which establish a maximum air temperature of 120^0F within the cavity. Thus, forced convection boundary conditions are established on all exposed surfaces within the reactor cavity. In addition, the forced convection establishes a reactor vessel insulation surface temperature of 150^0F. This results in radiant heat transfer from the reactor vessel insulation to the reactor support and inner surface of the shield wall. The outer surface of the shield wall is assumed to be cooled by natural convection of the 120^0F containment air. Radiation heating effects within the vertical members of the steel reactor supports near the elevation of the reactor core are also present. All of these effects were included in the boundary conditions for these two cases.

Figures 1 and 2 show the temperature distributions within the reactor beam-column assembly and primary shield wall respectively for a steady state normal operating condition. The slight discrepancy between the models at the concrete steel interface is caused by a concrete thermal capacitance difference in the two models and is of no significance to the stress analysis. Figure 3 shows the temperature distribution within the midplane of the reactor support foot of the hot leg. A reactor coolant temperature of 595^0F corresponding to operation at 100% power was assumed. The temperature distribution in Figure 1 is in good agreement with a scale model test of the reactor support cooling system.

For the startup case, several days are required for the concrete to heat up to a steady value from an initial ambient value of 70^0F. Furthermore, the horizontal support beam will not attain its steady state temperature distribution until the bulk of the surrounding concrete reaches an equilibrium value. This effect is illustrated in Figure 4 by the reactor beam-column assembly temperature distribution generated for the time of 18.0 hours after startup. Thus, a worse case is defined where the reactor beam-column assembly reaches its steady state value, but the concrete still contains large regions unaffected by the incoming heat flux. Therefore, the temperature distribution shown in Figure 5 is postulated as the worst startup situation for the primary shield wall.

D. LOSS OF FANS ACCIDENT

Assuming that both sets of fans (4 total) of the reactor cavity and support cooling systems should fail, LION indicates that a cavity air temperature of 420^0F would be attained within a 5.0 hour time interval. The basic assumptions made in this calculation are:

1. The air within the cavity does not mix with the rest of the air in the containment.

2. Natural convection and radiant heat transfer heat from the vessel insulation to the inner surface of the shield wall.

3. The initial air temperature is 120^0F.

The results of this study are shown in Figure 6. If the reactor operator detects the fan failure within 15 minutes of the occurrence, a shutdown of the reactor system can be initiated resulting in a reduction in the surface temperature of the reactor vessel and vessel insulation. The effect of such action on the hot leg coolant temperature is shown in Figure 7. The assumption is made that the core is in the End-Of-Life (EOL) stage and that only one charging pump is operating. Therefore, 1.75 hours are calculated to be needed to complete the boration cycle. Afterward the coolant temperature drops at a rate of 75^0F/Hr. The effect of such a shutdown on the cavity air temperature can be seen in the appropriate curve in Figure 6.

When these effects are imposed as time dependent boundary conditions on the reactor beam-column assembly and primary shield wall models for the long term operating and startup conditions, the temperature distributions shown in Figures 8, 9, and 10 are obtained.

The 4.5 hour time interval is chosen as the worst case because the reactor beam-column temperatures reach a maximum at this time. No distinction is made between the long term operating and startup cases for the reactor beam-column transients because the same initial temperature distribution is assumed for both situations. Thus, even with fast operator action, significant temperature increases in the reactor beam-column structures cannot be prevented. However, the low thermal conductivity of the concrete prohibits the effect from penetrating far into the shield wall.

E. LOSS OF COOLANT ACCIDENT (LOCA)

Following a LOCA, two different phases occur where severe stresses act on the reactor beam-column assembly and concrete shield wall. During the initial blowdown phase, the cavity pressure dominates the situation. However, thermal stresses are also present during this time interval. Peak cavity pressures occur during the first second. Even with the large Tagami steam condensation heat transfer coefficient, insignificant temperature increases occur during the 1.0 second interval. Therefore, the temperature distributions shown in Figures 11 and 12 are used for stress analysis during this time interval.

The second important phase occurs after a long time lapse and is known as the post-LOCA case when the cavity pressure is reduced, but thermal stresses dominate. In this situation the spillage water exists from the break (double ended hot leg slot) and begins to flood the cavity and sump drain. With all injection pumps operating a flow rate of 888 lbm/sec is attained. The cavity water level vs time is shown in Figure 14. Prior to 252 seconds the fluid exiting the break is predominantly hot steam. Afterward, water spillage is the main component of the fluid released via the break. A temperature of 200^0F for the leaking water can be shown to be conservative. As can be seen in Figure 13, this temperature is significantly lower than the cavity steam-air temperature. Thus as the water level in the cavity rises, the boundary condition on the reactor beam-column assembly changes locally from the Tagami correlation with the steam-air temperature to natural convection with the cooler spillage water. On the outer surface of the shield wall, a Tagami correlation for steam condensation is assumed. The results of imposing such boundary conditions on the models are shown in Figures 15 and 16. Although the steam-air temperature remains high, the combined effects of the cooling of the nozzle by ECCS flow and the spillage water alleviate the problem significantly.

REFERENCES

1. P.J. Holman, Heat Transfer, (McGraw-Hill Book Company) p. 176

2. Ibid, p. 229

3. F. Kreith, Principles of Heat Transfer (International Textbook Company) p. 314

4. Ibid, p. 343

5. Tagami, Takaski, "Interim Report on Safety Assessment and Facilities Establishment Project in Japan for Period Ending June 1965 (No. 1)"

6. H. Fujie, A. Yomanouchie, N. Sagawa, H. Ogasawara, T. Tagame "Studies for Safety Analysis of Loss-of-Coolant Accidents in Light-Water Power Reactors." Japan Atomic Energy Research Institute, NSJ-Tr112, Reactor Technology, March 1968.

TABLE 1

MATERIAL PROPERTIES

MATERIAL	STEEL	CONCRETE
DENSITY (1bm/ft^3)	490.0	150.0
SPECIFIC HEAT (BTU/1bm - ^{0}F)	0.11	0.2
THERMAL CONDUCTIVITY (BTU/hr-ft-^{0}F)	26.0	1.0

TABLE 2

STRUCTURAL SUPPORT BOUNDARY CONDITIONS

CASE	SURFACE	BOUNDARY CONDITIONS*	HEAT TRANSFER COEFFICIENT (BTU/HR-FT^2-$^\circ$F)	TEMPERATURE ($^\circ$F) (INITIAL)	TIME DEPENDENCE
NORMAL OPERATION (LONG TERM)	VERTICAL + HORIZONTAL BEAM SURFACES	FC[1]	0.85	120	NONE
	WITH VESSEL INSULATION	RT[2]	F$\sigma\,(T_1^4 - T_2^4)$ F = .24	150	
	WITH SURFACE OF SHIELD WALL	RT	F$\sigma\,(T_1^4 - T_2^4)$ F = .91	125	
	REGION OF STRUCTURAL COOLING SYSTEM	FC[3]	9 - 12	120	
	INNER SURFACE OF HOT LEG PIPING	FT	- - -	595	
	INTERIOR OF CAVITY FLOOR AND WALLS	FT	- - -	120	
STARTUP	VERTICAL + HORIZONTAL BEAM SURFACES	FC	0.85	70	LINEAR INCREASE TO 120°F in 5.0 HRS.

TABLE 2 Cont'd

CASE	SURFACE	BOUNDARY CONDITIONS*	HEAT TRANSFER COEFFICIENT (BTU/HR-FT^2-^{0}F)	TEMPERATURE (^{0}F) (INITIAL)	TIME DEPENDENCE
STARTUP (Cont'd)	WITH VESSEL INSULATION	RT	F$\sigma(T_1^4 - T_2^4)$ F=.24	70	LINEAR INCREASE TO 150^{0}F IN 5.0 HOURS
	WITH SURFACE OF SHIELD WALL	RT	F$\sigma(T_1^4 - T_2^4)$ F=.91	70	LINEAR INCREASE TO 125^{0}F IN 5.0 HOURS
	REGION OF STRUCTURAL COOLING SYSTEM	FC	9-12	70	LINEAR INCREASE TO 120^{0}F IN 5.0 HOURS
	INNER SURFACE OF HOT LEG PIPING	FT	- - - -	70	LINEAR INCREASE TO 595^{0}F IN 5.0 HOURS
	INTERIOR OF CAVITY FLOOR AND WALLS	FT	- - - -	70	NONE
LOSS OF FANS (LONG TERM AND DURING STARTUP)	VERTICAL + HORIZONTAL BEAM SURFACES	NC^4	$.18\,\Delta T^{1/3}$	120	SEE FIGURE B-6
	WITH VESSEL INSULATION	NC	F$\sigma(T_1^4 - T_2^4)$ F=.24	150	AS OBTAINED FROM MODEL DESCRIBED IN SECTION B-6
	WITH SURFACE OF SHIELD WALL	RT	F$\sigma(T_1^4 - T_2^4)$	125	SAME AS ABOVE

TABLE 2 Cont'd

CASE	SURFACE	BOUNDARY CONDITIONS*	HEAT TRANSFER COEFFICIENT (BTU/HR-FT^2-$^\circ$F)	TEMPERATURE ($^\circ$F) (INITIAL)	TIME DEPENDENCE
LOSS OF FANS (LONG TERM AND DURING STARTUP) (Cont'd)	INNER SURFACE OF HOT LEG PIPING	FT	- - - -	120	SEE FIGURE B-7
	INTERIOR OF CAVITY FLOOR AND WALLS	FT	- - - -	120	NONE
LOCA AND POST-LOCA	VERTICAL + HORIZONTAL BEAM SURFACES ABOVE WATER	SC[5,6]	TAGAMI COEFFICIENT	120	SEE FIGURE B-13, B-14
	VERTICAL + HORIZONTAL BEAM SURFACES SUBMERGED	NC	70 $\Delta T^{1/3}$	120	SEE FIGURE B-13, B-14
	INNER SURFACE OF PIPING	FT	- - - -	595	SEE FIGURE B-13
	INTERIOR OF CAVITY FLOOR AND WALLS	FT	- - - -	120	NONE

* BOUNDARY CONDITIONS:
FC FORCED CONVECTION
NC NATURAL CONVECTION
SC STEAM CONDENSATION
RT RADIANT HEAT TRANSFER
FT FIXED TEMPERATURE

TABLE 3

CONCRETE SHIELD WALL BOUNDARY CONDITIONS

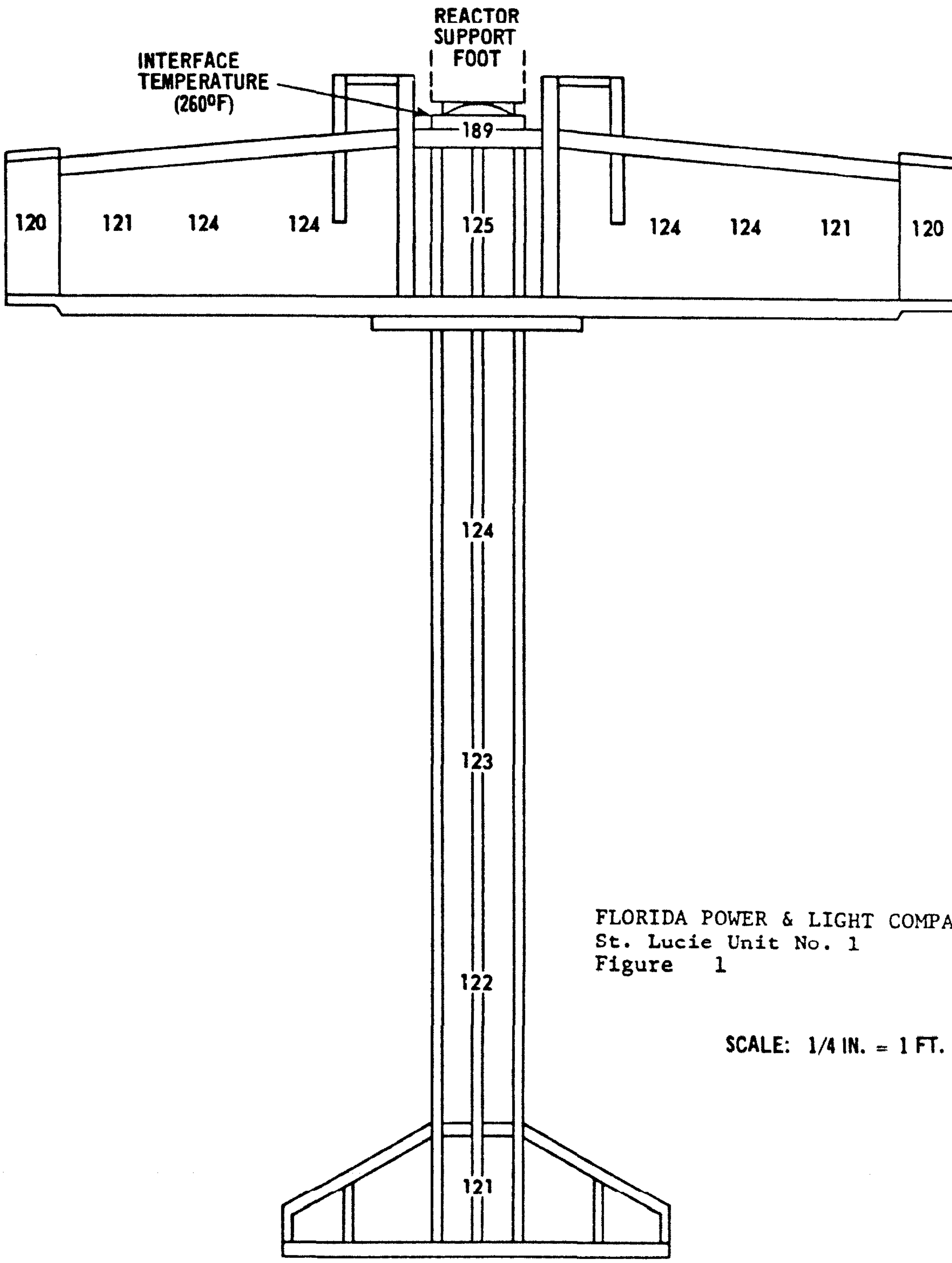
CASE	SURFACE	BOUNDARY CONDITIONS*	HEAT TRANSFER COEFFICIENT (BTU/HR-FT^2-^{0}F)	TEMPERATURE (^{0}F) (INITIAL)	TIME DEPENDENCE
NORMAL OPERATION	INNER SURFACE	FC[1]	0.85	120	NONE
	OUTER SURFACE	NC[4]	$.18\ \Delta T^{1/3}$	120	
	WITH VESSEL INSULATION	RT[2]	$F\sigma(T_1^4 - T_2^4)$ F=.24	150	
	INTERFACE WITH REACTOR SUPPORT	FT	- - - -	AS OBTAINED FROM STRUCTURAL SUPPORT MODEL	
STARTUP	INNER SURFACE	FC	0.85	70	LINEAR INCREASE TO 120^{O}F IN 5.0 HOURS
	OUTER SURFACE	NC	$.18\ \Delta T^{1/3}$	70	LINEAR INCREASE TO 120^{O}F IN 5.0 HOURS
	WITH VESSEL INSULATION	RT	$F\sigma(T_1^4 - T_2^4)$ F=.24	70	LINEAR INCREASE TO 150^{O}F IN 5.0 HOURS
	INTERFACE WITH REACTOR SUPPORT	FT	- - - -	70	AS OBTAINED FROM STRUCTURAL SUP-PORT MODEL

TABLE 3 Cont'd

CASE	SURFACE	BOUNDARY CONDITIONS*	HEAT TRANSFER COEFFICIENT (BTU/HR-FT^2-^{O}F)	TEMPERATURE (^{O}F) (INITIAL)	TIME DEPENDENCE
LOSS OF (LONG TERM + STARTUP)	INNER SURFACE	NC	.18 $\Delta T^{1/3}$	120	SEE FIGURE B-6
	OUTER SURFACE	NC	.18 $\Delta T^{1/3}$	120	NONE
	WITH VESSEL INSULATION	RT	$F\sigma(T_1^4 - T_2^4)$	150	AS OBTAINED FROM MODEL DESCRIBED IN SECTION B-6
	INTERFACE WITH REACTOR SUPPORT	FT	- - - -	AS OBTAINED FROM STRUCTURAL SUPPORT MODEL	AS OBTAINED FROM STRUCTURAL SUPPORT MODEL
LOCA + POST-LOCA	INNER SURFACE	SC + [4,5,6] NC (WHEN SUBMERGED UNDER WATER)	TAGAMI COEFFICIENT + .70 $\Delta T^{1/3}$	120	SEE FIGURE B-13 B-14
	OUTER SURFACE	SC [5,6]	TAGAMI COEFFICIENT	120	SEE FIGURE B-13
	INTERFACE WITH REACTOR SUPPORT	FT	- - - -	AS OBTAINED FROM STRUCTURAL SUPPORT MODEL	AS OBTAINED FROM STRUCTURAL SUPPORT MODEL

*FC FORCED CONVECTION
NC NATURAL CONVECTION
SC STEAM CONDENSATION
RT RADIANT HEAT TRANSFER
FT FIXED TEMPERATURE

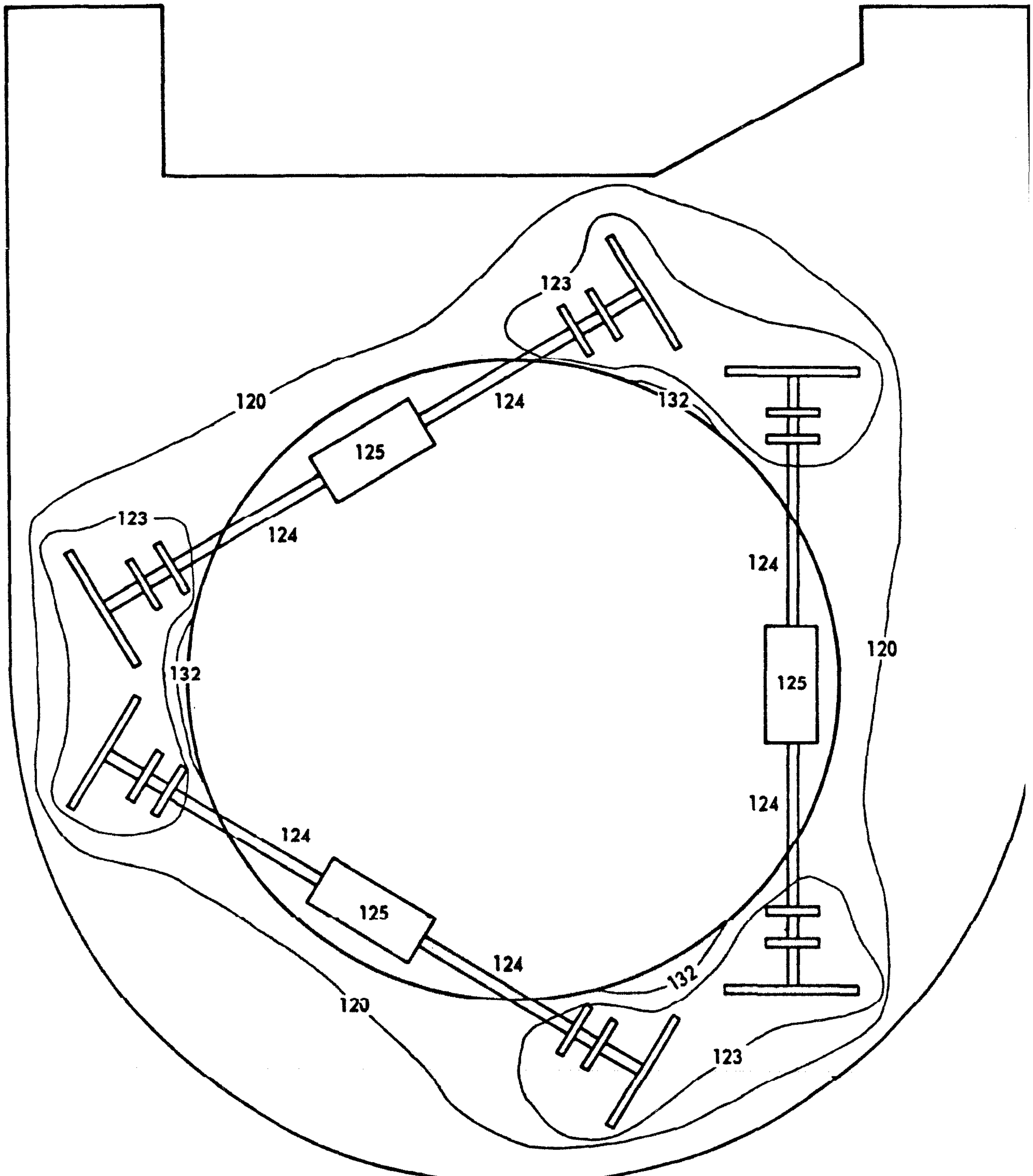
REACTOR SUPPORT TEMPERATURE DISTRIBUTION
STEADY STATE NORMAL OPERATION CONDITIONS
AIR TEMPERATURE = 120°F



FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 1

SCALE: 1/4 IN. = 1 FT.

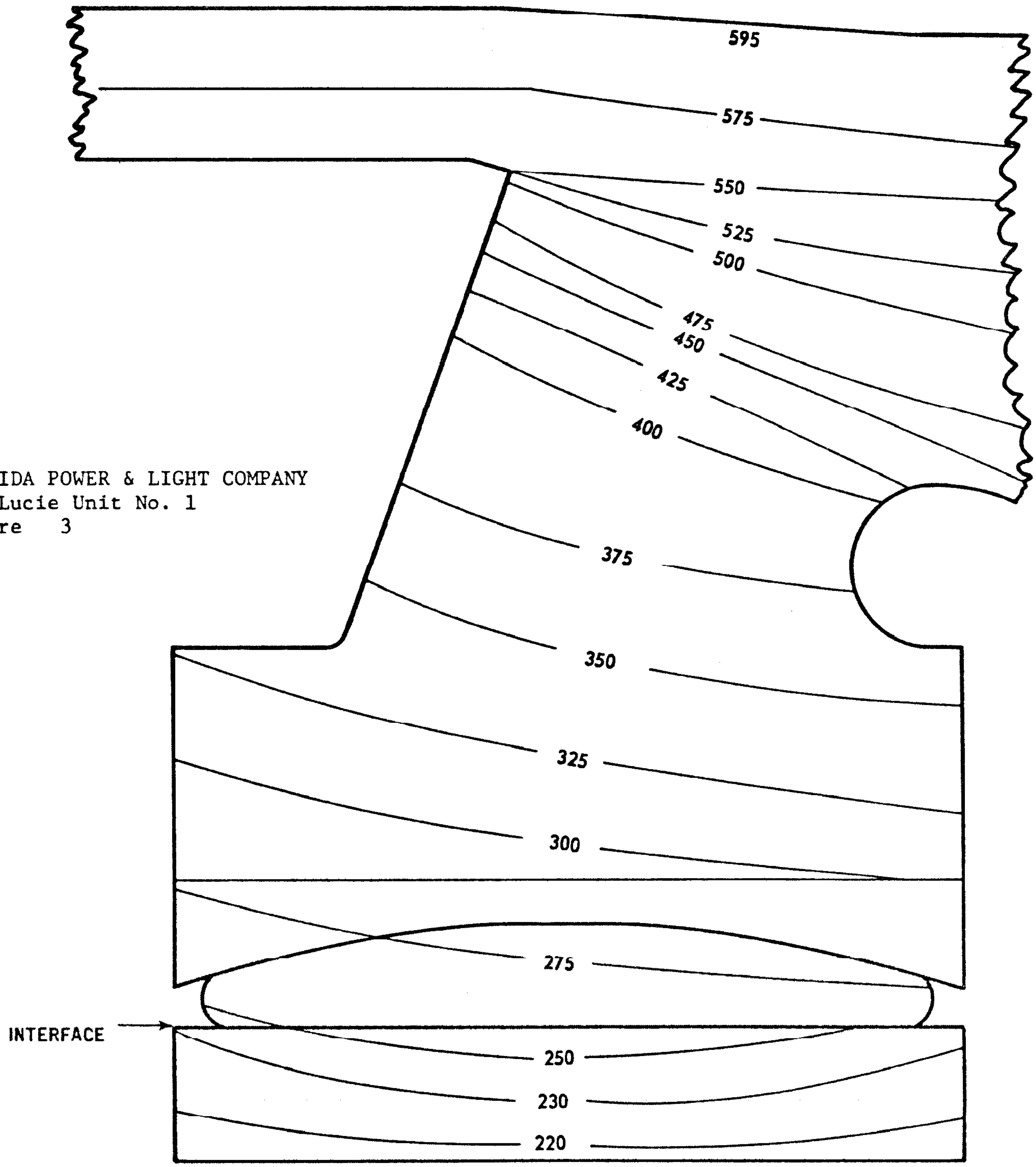
CONCRETE SHIELD WALL ELEVATION 22.5 FEET
STEADY STATE TEMPERATURE DISTRIBUTION
NORMAL OPERATION AIR TEMPERATURE = 120°F



'LORIDA POWER & LIGHT COMPANY
.t. Lucie Unit No. 1
'igure 2

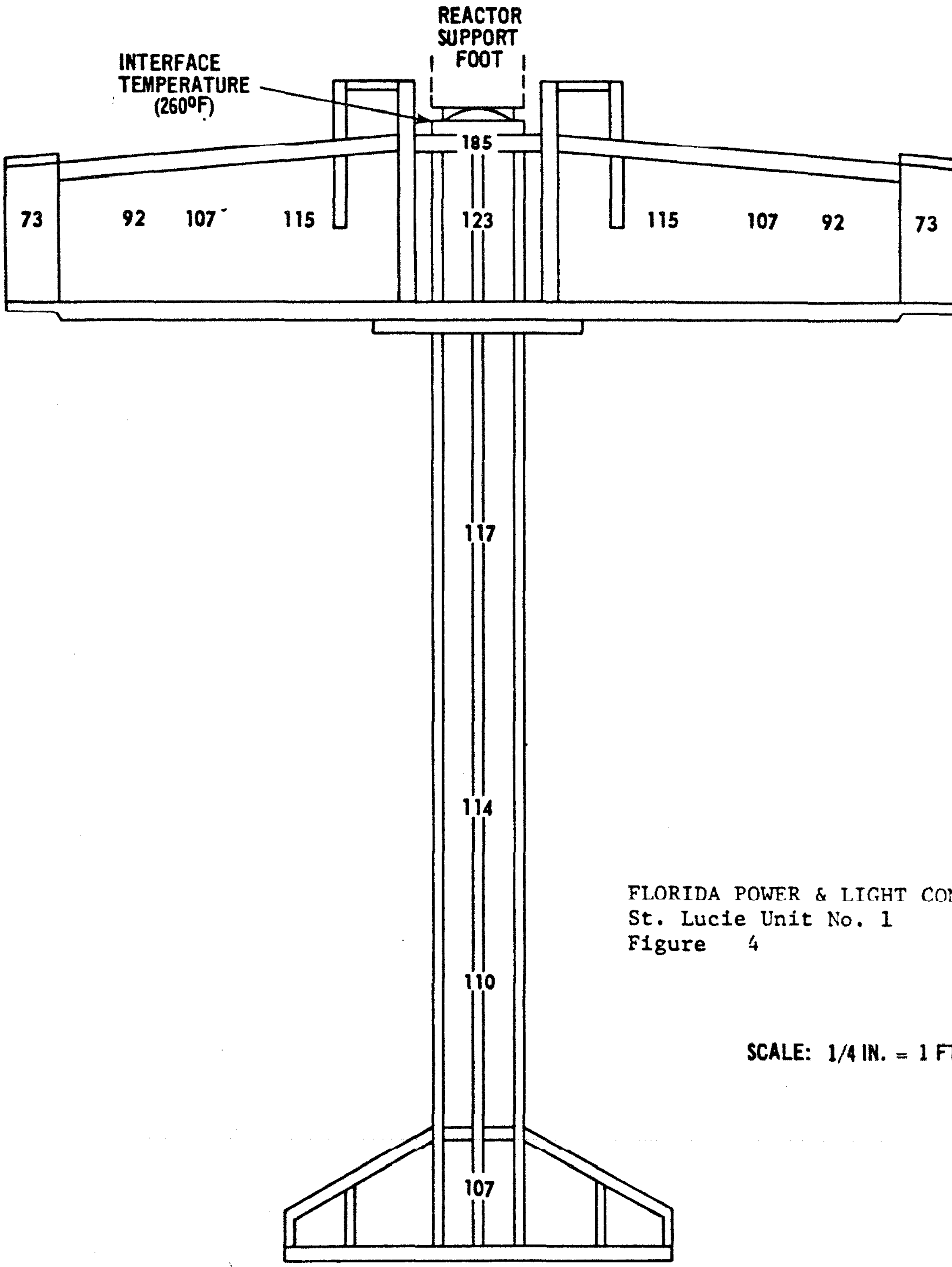
SCALE: 3/16 IN. = 1 FT.

HOT LEG STEADY STATE NOZZLE AND SUPPORT FOOT TEMPERATURE DISTRIBUTION



FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 3

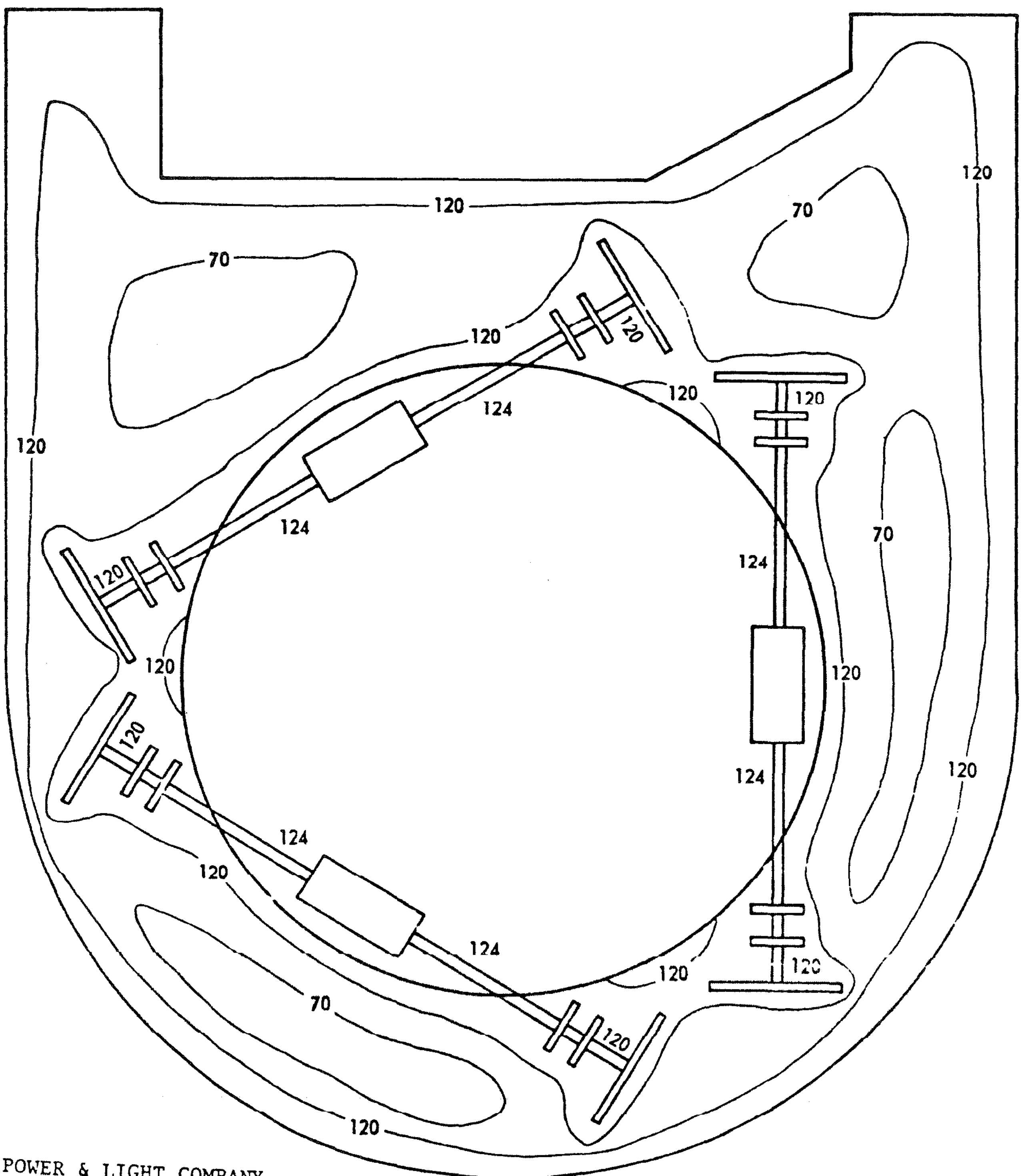
REACTOR SUPPORT TEMPERATURE DISTRIBUTION
START-UP TIME = 18.0 HOURS
AIR TEMPERATURE = 120°F
INITIAL TEMPERATURE = 70°F



FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 4

SCALE: 1/4 IN. = 1 FT.

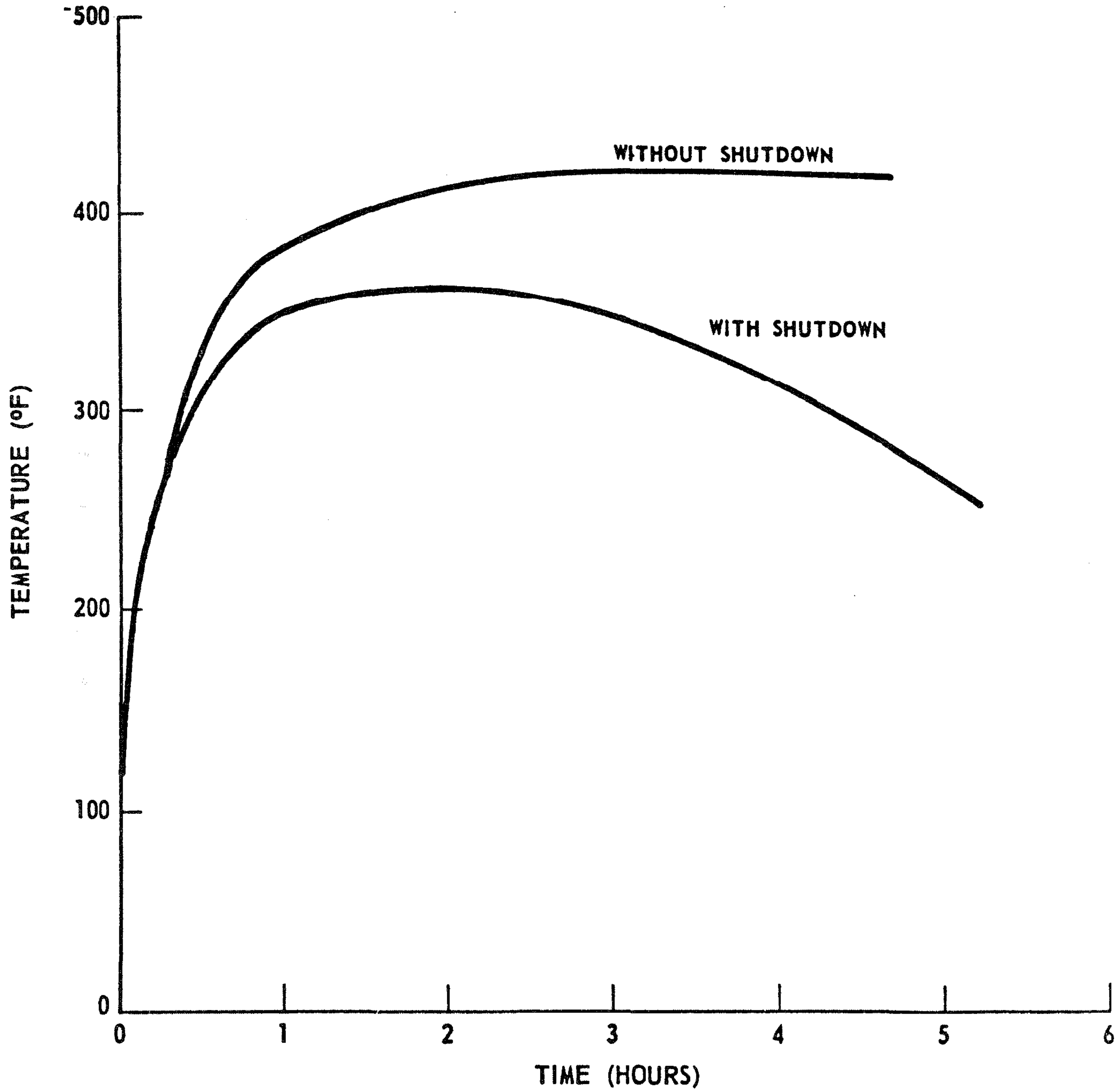
CONCRETE SHIELD WALL ELEVATION 22.5 FEET
TEMPERATURE DISTRIBUTION
SEVERAL DAYS AFTER START-UP



..ORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 5

SCALE: 3/16 IN. = 1 FT.

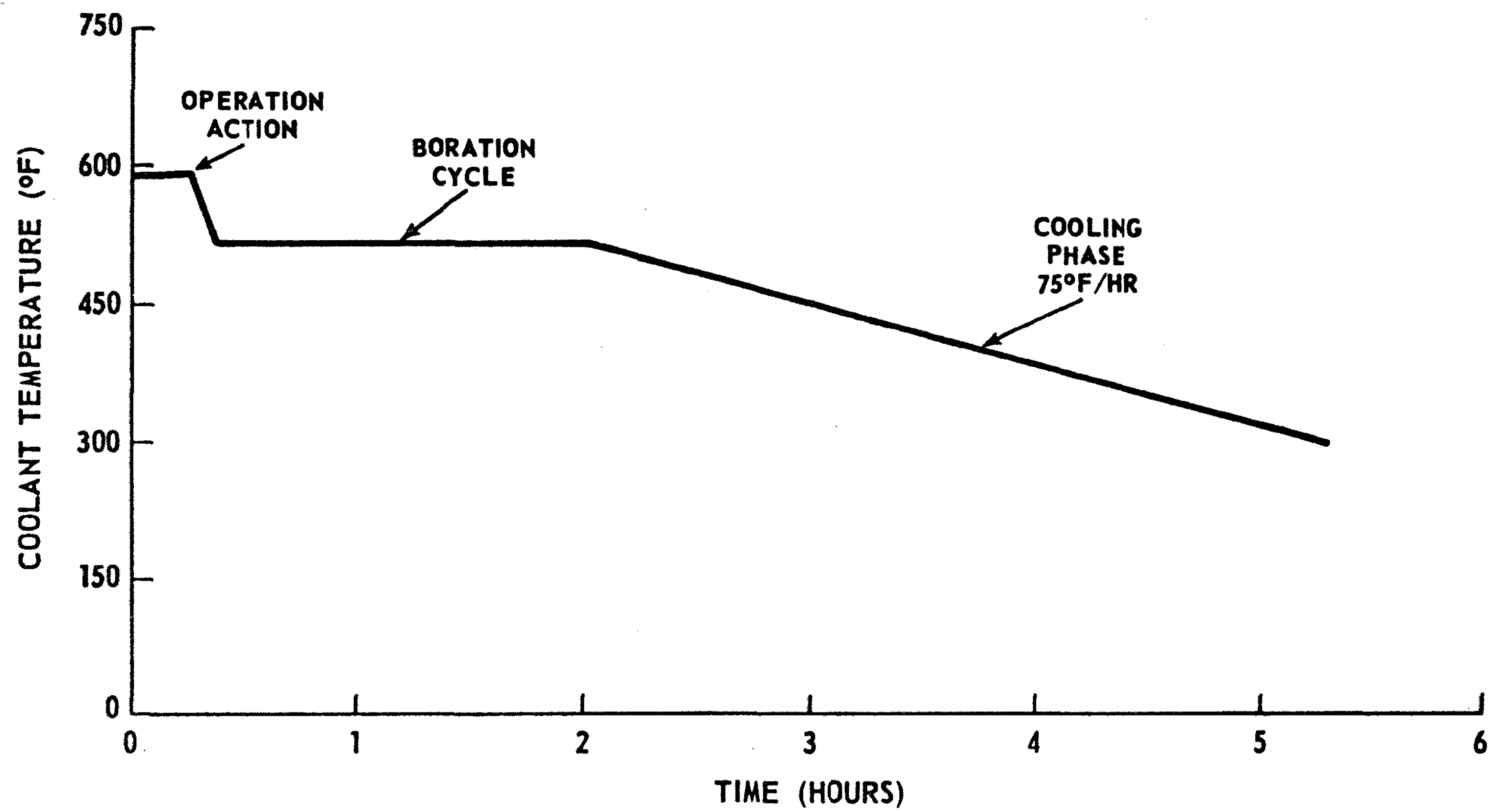
MEAN AIR TEMPERATURE OF REACTOR CAVITY
FOLLOWING LOSS OF FANS ACCIDENT



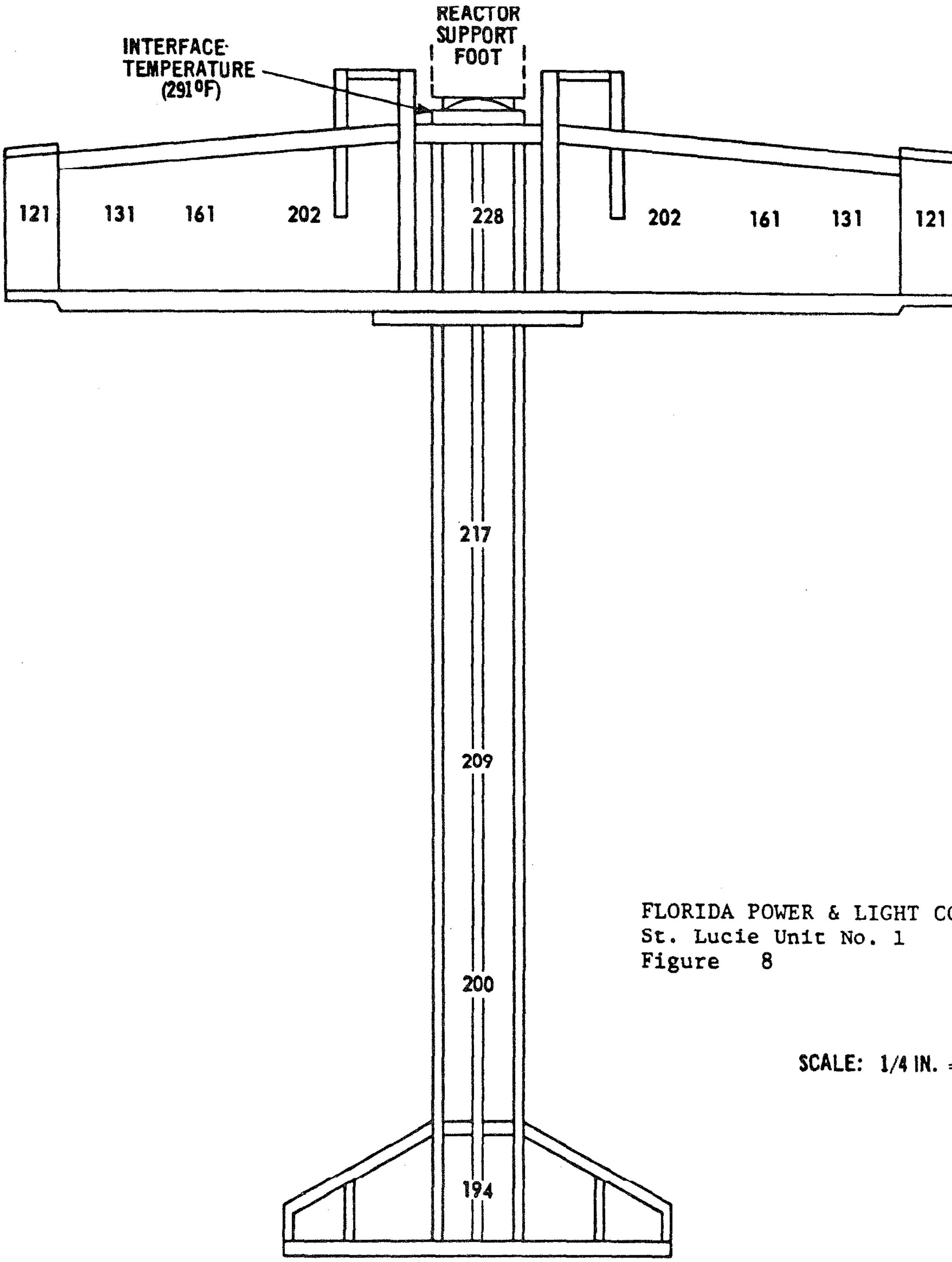
FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 6

FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 7

REACTOR COOLANT TEMPERATURE VERSUS
TIME FOLLOWING OPERATOR ACTION
FOR LOSS OF FANS CASE



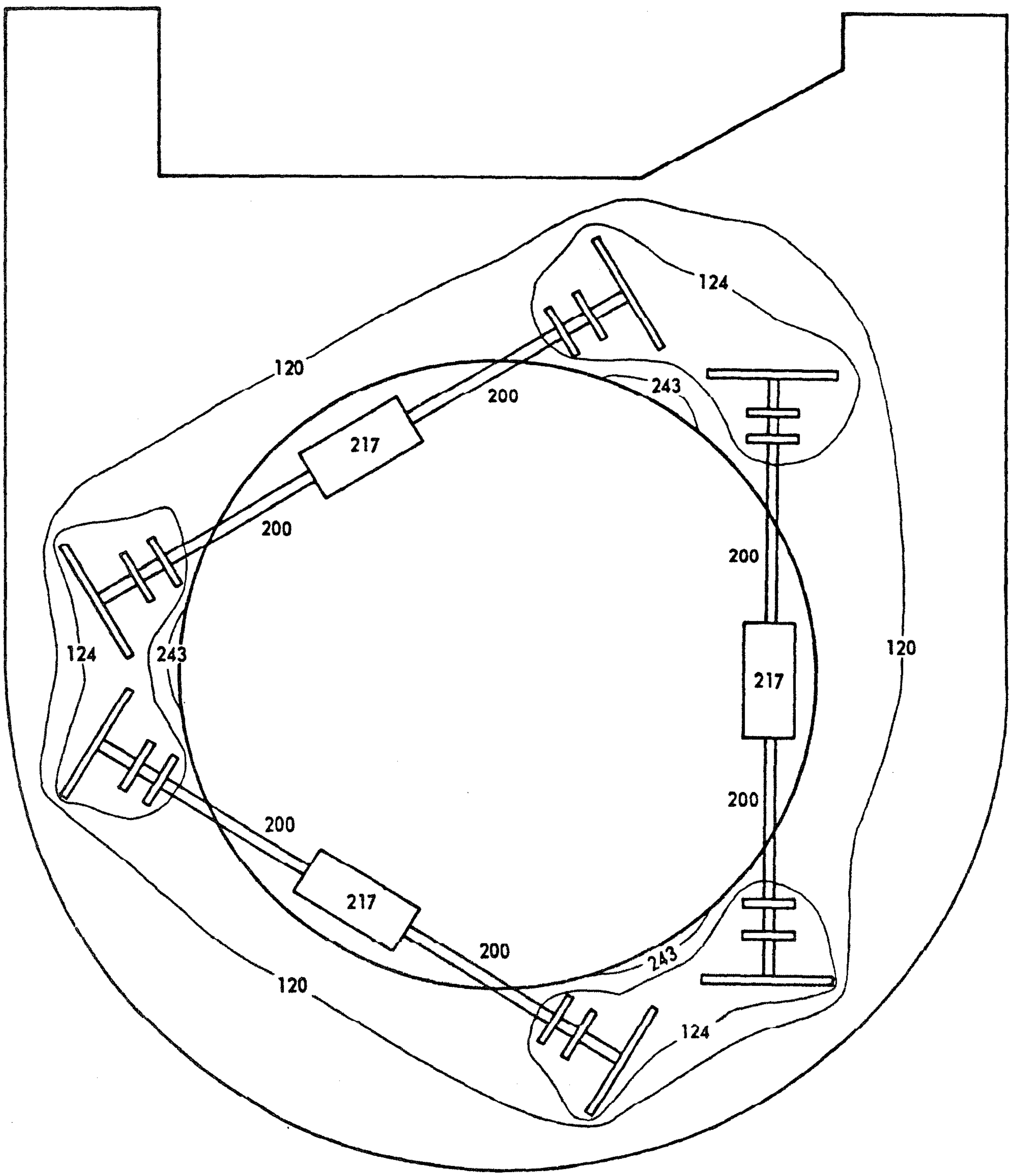
REACTOR SUPPORT TEMPERATURE DISTRIBUTION
LONG TERM LOSS OF FANS TIME = 4.5 HOURS
AIR TEMPERATURE = 290°F



FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 8

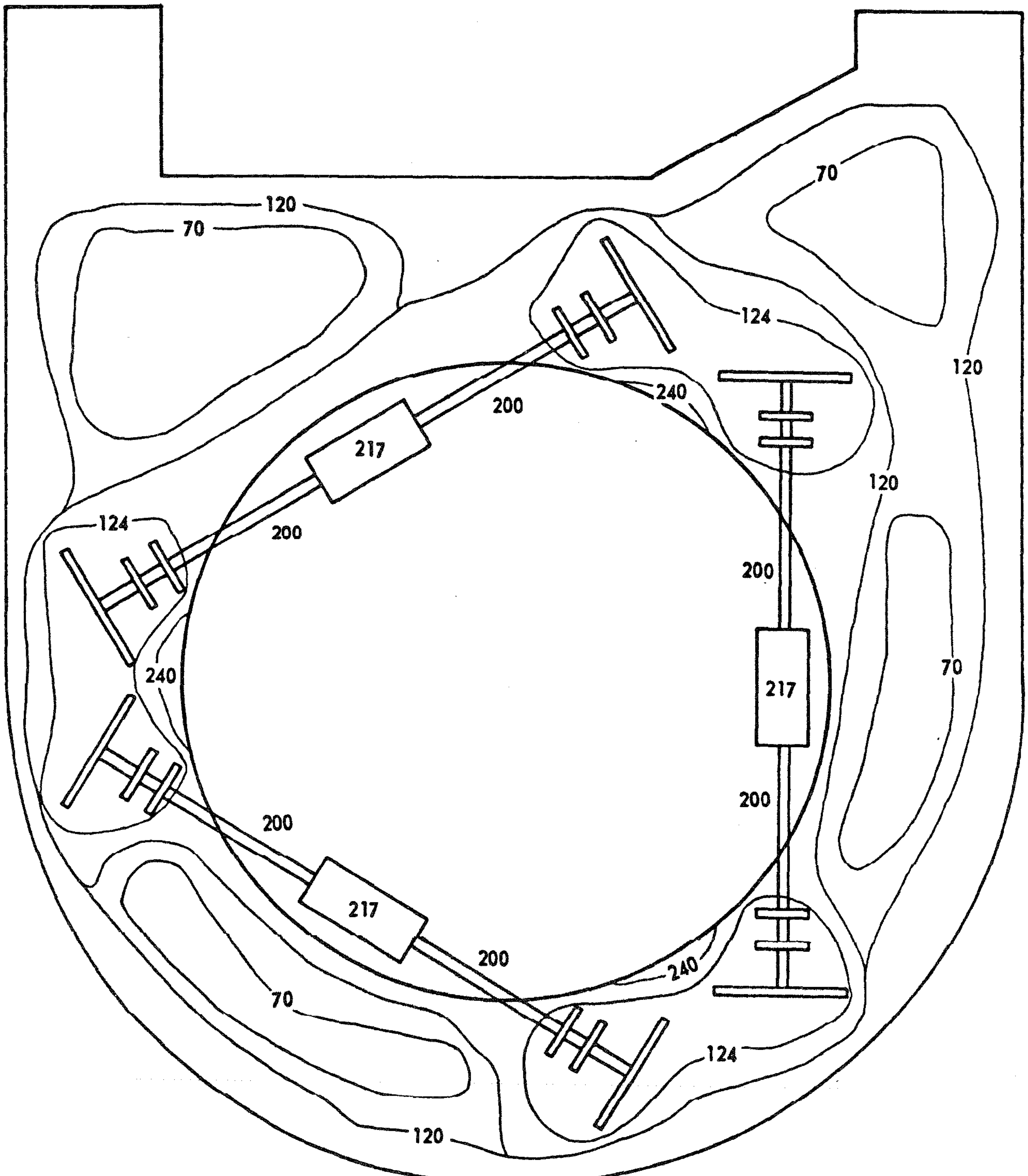
SCALE: 1/4 IN. = 1 FT.

CONCRETE SHIELD WALL ELEVATION 22.5 FEET
TEMPERATURE DISTRIBUTION
4.5 HOURS AFTER LOSS OF FANS
OCCURRENCE DURING NORMAL OPERATIONS



SCALE: 3/16 IN. = 1 FT.

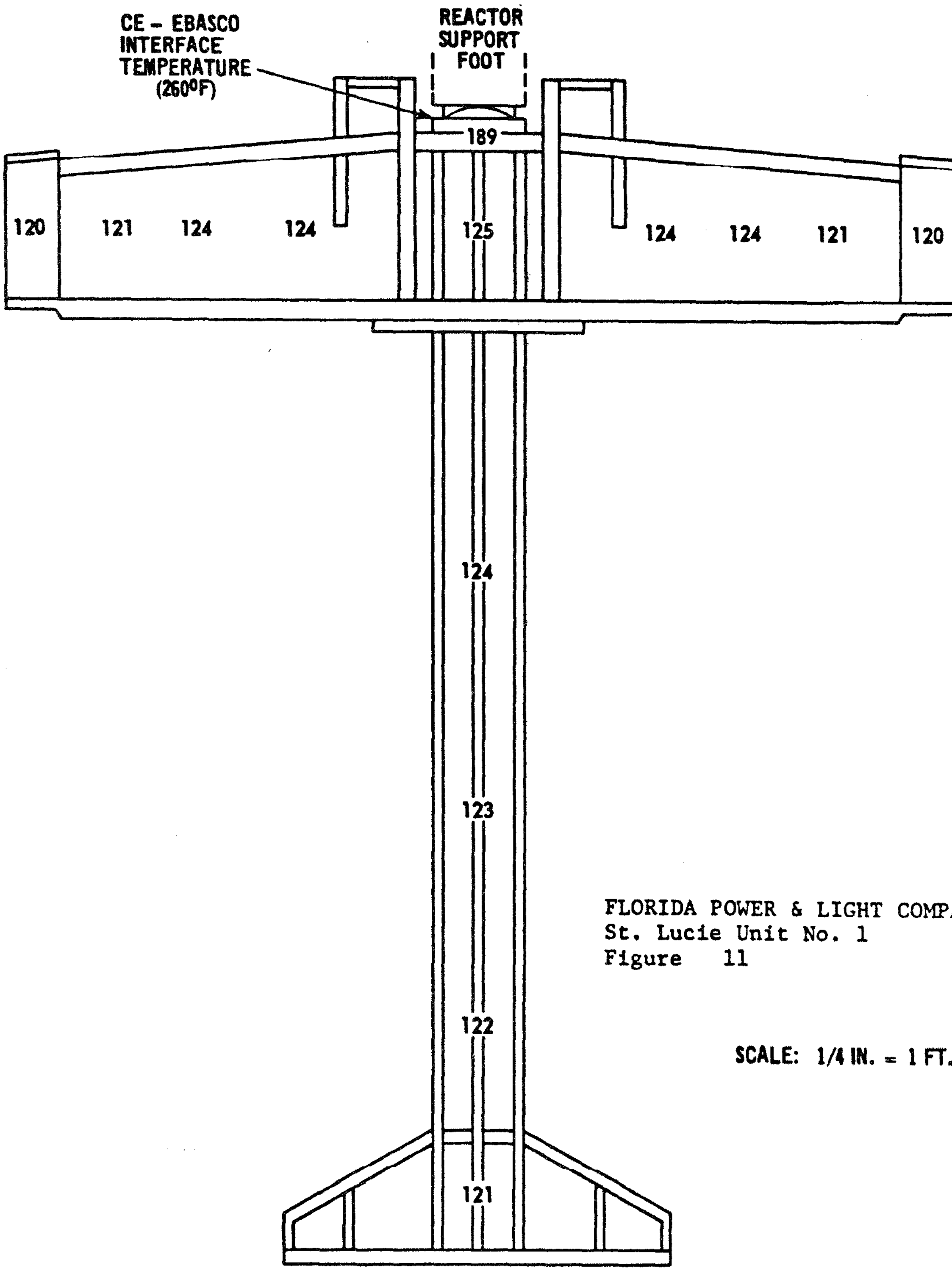
CONCRETE SHIELD WALL ELEVATION 22.5 FEET
TEMPERATURE DISTRIBUTION
4.5 HOURS AFTER LOSS OF FANS
OCCURRENCE DURING START-UP



FLORIDA POWER & LIGHT COMPANY
ST. Lucie Unit No. 1
Figure 10

SCALE: 3/16 IN. = 1 FT.

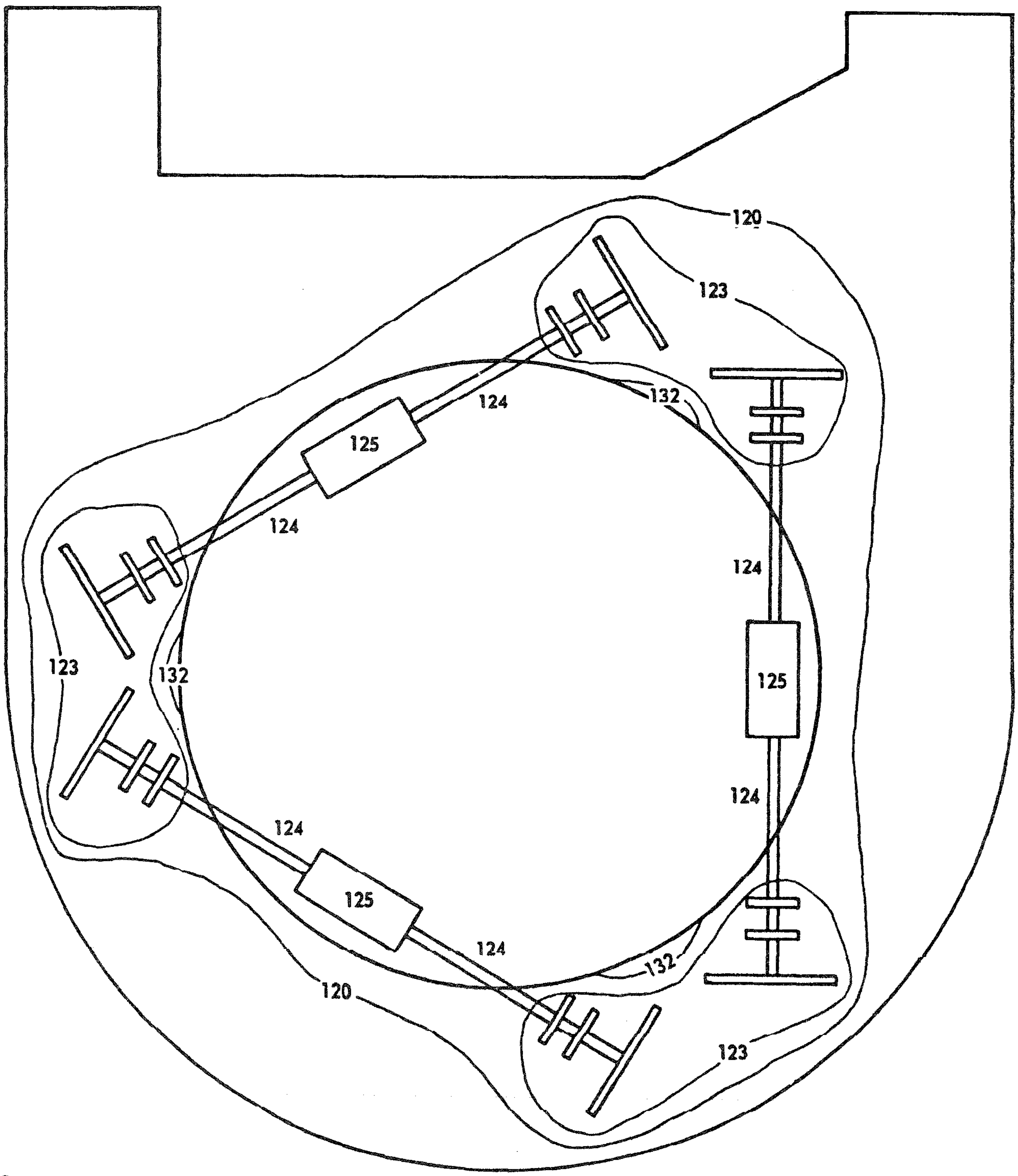
REACTOR SUPPORT TEMPERATURE DISTRIBUTION
LOCA TIME = 1.0 SECOND
AIR TEMPERATURE - 120°F



FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 11

SCALE: 1/4 IN. = 1 FT.

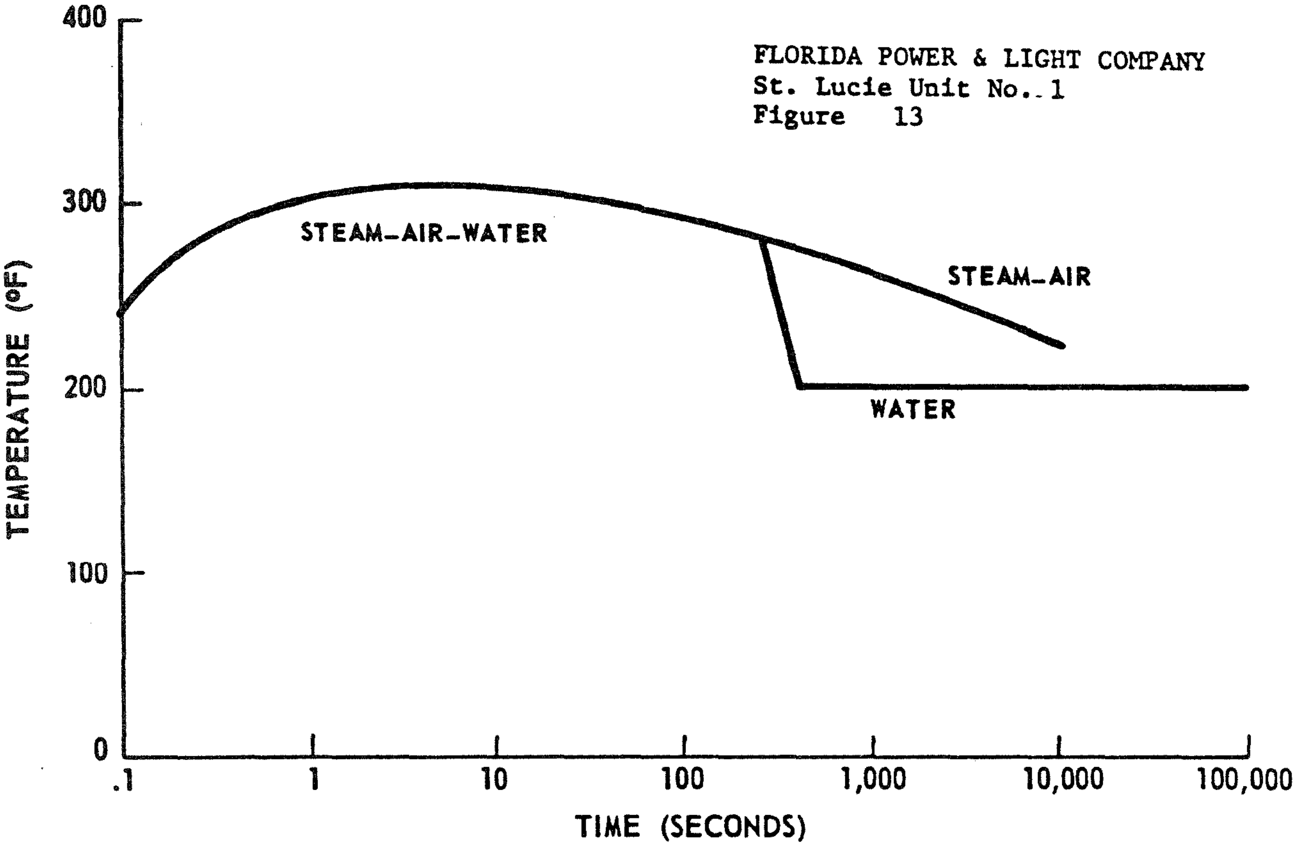
CONCRETE SHIELD WALL ELEVATION 22.5 FEET
LOCA TEMPERATURE DISTRIBUTION TIME = 1.0 SECOND



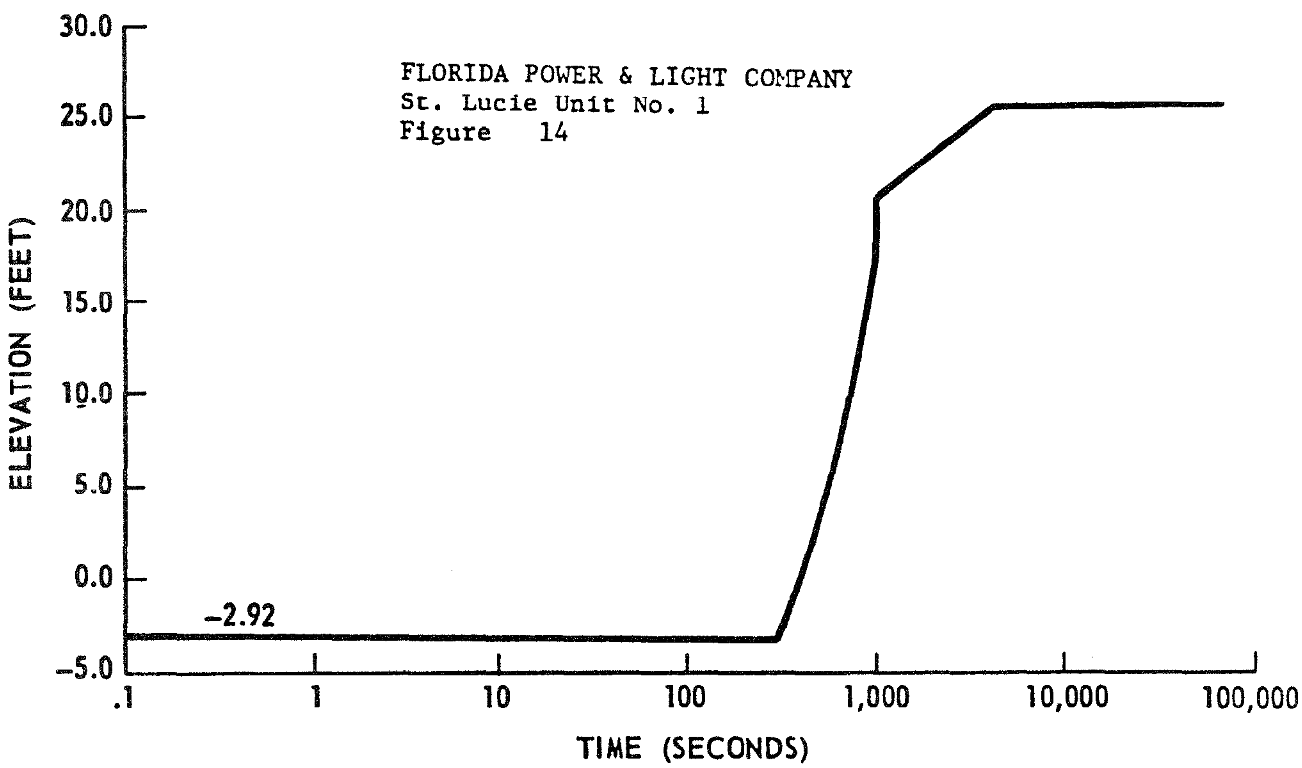
FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 12

SCALE: 3/16 IN. = 1 FT.

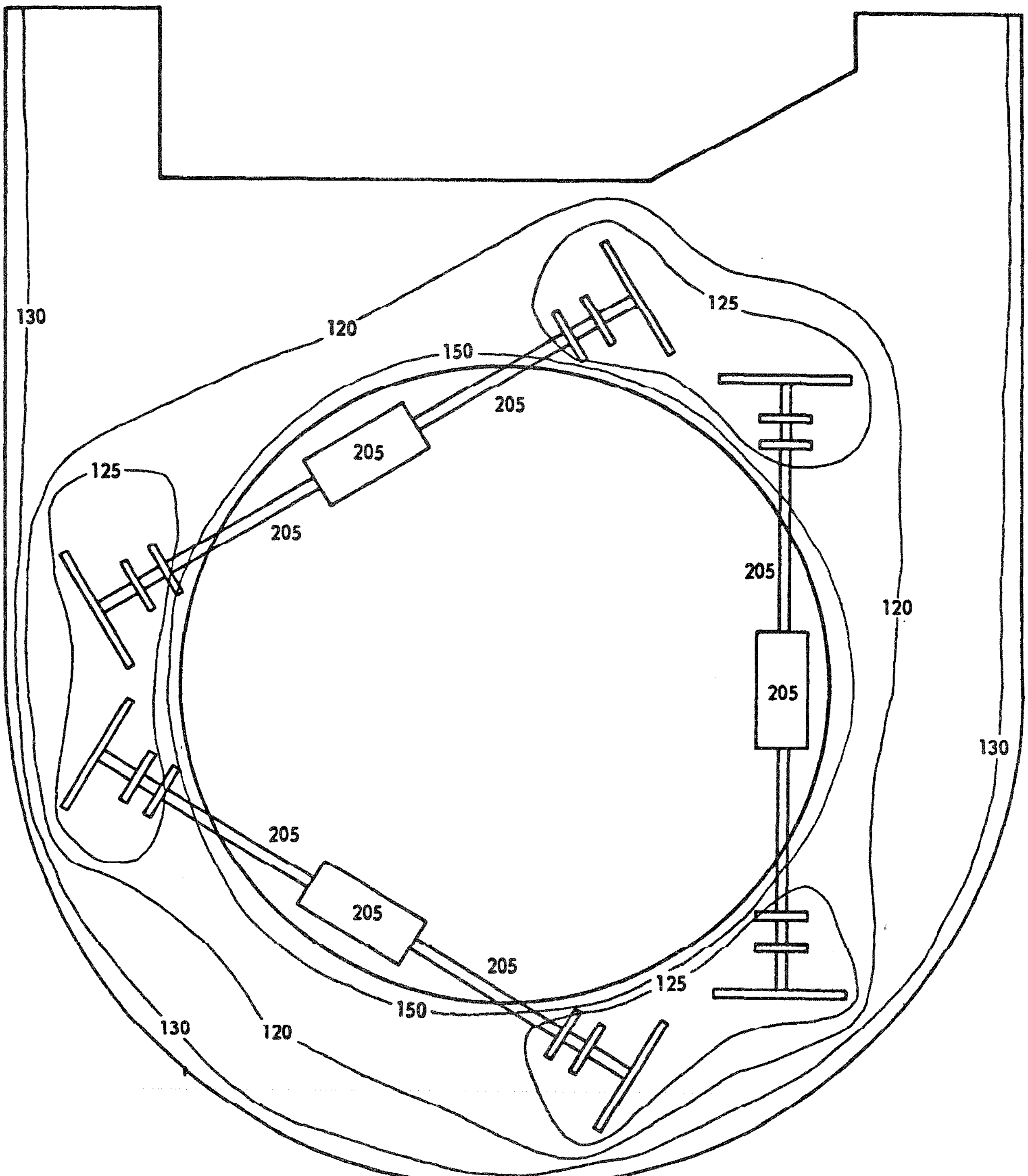
POST-LOCA BOUNDARY CONDITIONS



REFLOOD WATER LEVEL VERSUS TIME



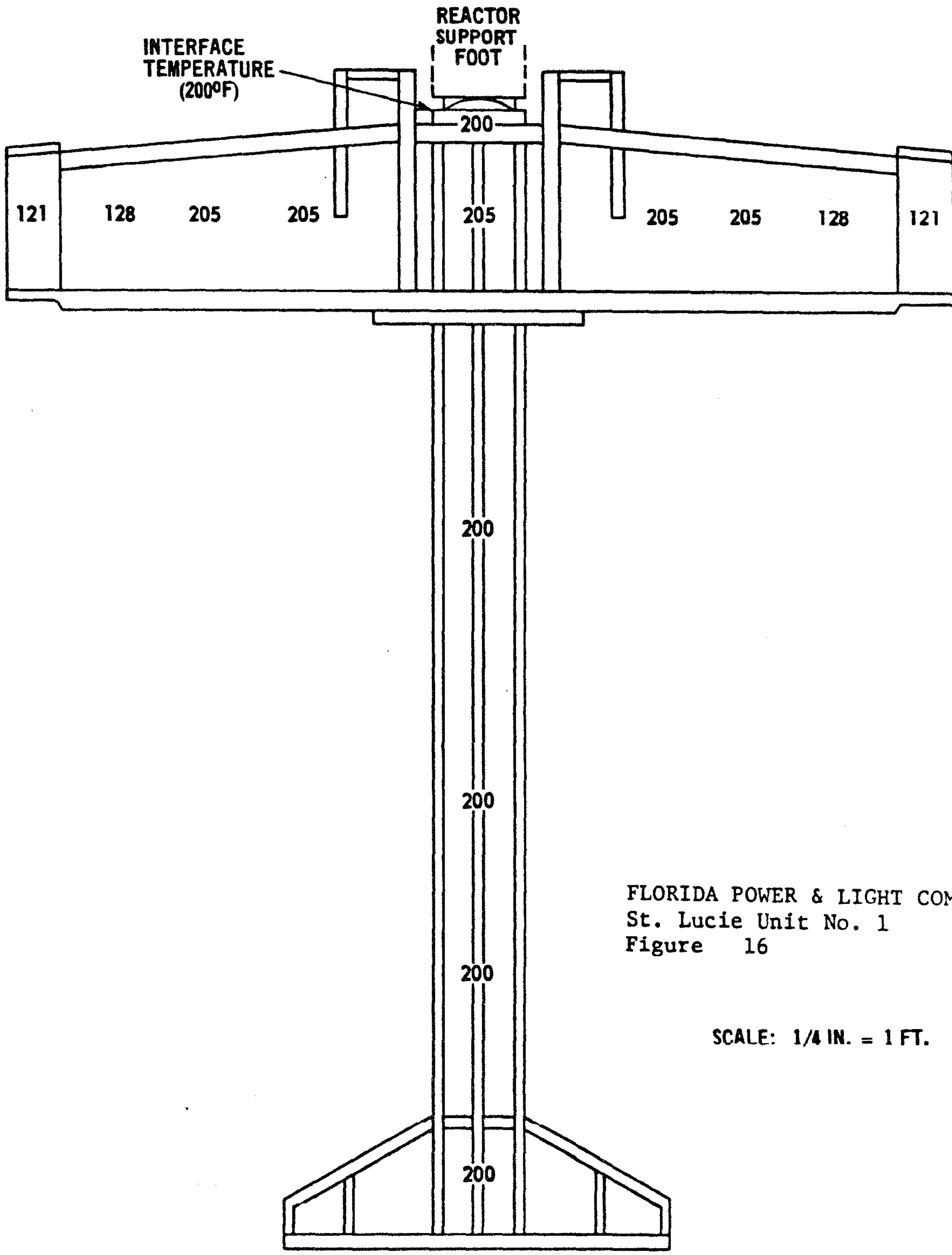
CONCRETE SHIELD WALL ELEVATION 22.5 FEET
POST-LOCA TEMPERATURE DISTRIBUTION
WITH NORMAL OPERATION AS INITIAL
CONDITION TIME = 1.0 HOUR



FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 15

SCALE: 3/16 IN. = 1 FT.

REACTOR SUPPORT TEMPERATURE DISTRIBUTION
POST-LOCA TIME = 1.0 HOUR



FLORIDA POWER & LIGHT COMPANY
St. Lucie Unit No. 1
Figure 16

SCALE: 1/4 IN. = 1 FT.

ATTACHMENT 2

ALFRED L. PARME
CONSULTING ENGINEER
6787 AVENIDA ANDORRA, LA JOLLA, CALIFORNIA 92037
459-6877

January 24, 1975

Mr. Anthony Ferlito
Ebasco Services, Inc.
2 Rector St.
New York, New York 10006

Dear Mr. Ferlito:

In conformance with our understanding, I have reviewed the design procedure applied to the analysis of the supporting steel beams of the nuclear reactor for the Hutchinson Island Plant of the Florida Power Plant regarding the effect of various load and thermal conditions. In addition to extended discussions with your staff, the drawings were studied and approximate calculations were made to confirm the range of stress values recorded by your staff.

The procedure employed by your staff incorporates the most sophisticated tool, finite element, now available in structural design. The basic assumptions made with regard to the properties of material are sound and will lead to conservative answers.

The stresses induced, tensile and compressive due to thermal changes and external load are well within the strength capacity of the concrete. As a matter of fact, the long time response of the concrete to the thermal changes will be to decrease the stresses below those shown. The magnitude of the displacements are so small that slight creep in the concrete will decrease greatly the stresses induced by thermal effects.

Although, in my opinion, the cylinder will be able to resist by its concrete shear resistance the forces generated by the thermal expansion of the steel beam, there is an additional safety factor. The vertical steel will act through shear friction behavior as an additional response mechanism. Thus the capacity of the concrete cylinder surrounding the steel beams is augmented over that which can be implied by an evaluation of the stress level.

The elastic response of the concrete cylinder to thermal expansion of the steel beams will cause the stresses induced in the beam to be approximately 50% of those based on the assumption that the beam is fully fixed at the ends.

Mr. Anthony Ferlito January 24, 1975

External lateral loads applied to the steel beams will be resisted primarily by the concrete cylinder. The steel beams will act primarily as a tie or strut, or combination of the two to transmit the load to the concrete. There will be slight lateral bending in the beams due to interaction between the steel beam and concrete. This bending should be minor due to the relative stiffness of the concrete circular ring and steel beam.

Sincerely yours,

Alfred L. Parme

Alfred L. Parme

ALP:ab
encl.

PROFESSIONAL RECORD

Name: Alfred L. Parme   Tele: 714-459-6877

Address: 6787 Avenida Andorra
La Jolla, California 92037

Education: Cornell University (3 years) 1932 - 1935 C.E. 1935

Honors: Tau Beta Pi; Chi Epsilon, McMillan Scholarship
Illinois Institute of Technology - 1940-1942 -Advanced courses on Structural Design.

Professional Honors: 1953 - Fuertes Medal by Cornell University for ASCE Manual #31 Design of Cylindrical Concrete Shells.

1959 - Moisseif Medal by ASCE (American Society of Civil Engineers) for contribution on design of shells of double curvature.

1964 - Lindau Award by the American Concrete Institute for outstanding contribution on design of reinforced concrete.

1969 - Rickey Medal by ASCE for paper, "Arch Dam Layout Facilitated by Computer Usage."

1972 - Martin P. Korn Award by Prestressed Concrete Institute for "American Practice in Seismic Design"

- Vice-President of International Association of shell structures

1974 - Elected to the National Academy of Engineering.

- Member of ACI committees on Columns; Ultimate Strength Design; Limit Design (former Chairman); Shell Structures (former Chairman); Concrete Joints and Seismic Design.

Structural Consultant to:

EBASCO SERVICES, Inc. (Arch Dams)
Harza Engineering Company (Arch Dams)
Gulf General Atomic (Nuclear Reactor)
Atlantic Richfield Oil Company (Waste Tanks)
Wiss Janney & Associates (Tunnels)
Worthington Corp.(Gas Turbine Generating Plants)
Lev Zetlin (Hangar)
Wayman C Wing (Hi-rise Bldg)
Edo Belli (Schools & Churches)
W Wenzler (Diversified Structures)
Construction Ministry of Venezuela (Seismic Design)
D.S.I. of Turkey (Arch Dam)
Prestressing Concrete Institute (Handbook)
Concrete Reinforcing Steel Institute (Handbook)

Employment Record:

1968 - 1969	Partner, Office of J. Fruchtbaum Associates, 1965 Sheridan Drive, Buffalo, N.Y. 14223 - Tele: 716-877-3350. Specialized in industrial buildings and unusual projects as MIT Nuclear Reactor and Cornell Aeronautical Bldg.
1940 - 1968	Portland Cement Association, Old Orchard Road, Skokie, Illinois, 60076 - Supervisor J. D. Piper.
1960 - 1968	Director of Advanced Engineering Dept. As head of this group was responsible for maintaining PCA leadership in structural design. In this capacity applied advanced engineering, physical principles and test data to secure efficient use of concrete. New methods of design were developed for arch dams, curved concrete shells, building frames and prestressed concrete. Many of the contributions have become classics in the field. In the eight years of its existence, this small selected group published 21 Advanced Engineering Bulletins, in addition to providing aid for other departments. Also, acted as consultant on many unusual projects.
1957 - 1960	Manager of Structural Bureau - In charge of 20 man department responsible for structural promotion; duties similar to the above, but on less technical level and interspersed with promotional activity.
1940- 1957	Structural Engineer - In charge of technical publication R/C and development of structural design techniques. Likewise, assisted engineers on unusual structures which required knowledge beyond average practicing engineer. Contribution of original nature made on design of storage tanks (circular or rectangular) rigid frame bridges, skewed arch bridges and domes.
Oct. 1,1952 to May 1, 1953	Overseas Consultant Incorporated - 2 Rector St., N.Y.C. Title: Consulting Engineer Duties: Responsible for design of first arch dam to be built in Japan and training of Japanese engineers on design of training of Japanese engineers on design of arch dam. Investigation of foundation yielding and associated problems with arch dams.
Sept.1, 1943 to Oct. 1, 1945	The Republic Aviation Corp., Farmingdale, N.Y. Title: Senior Stress Analyst Supervisor: Frank Mevay, Head of Stress Analysis Group Duties: As a Senior Stress Analyst, my main function was to supervise the analysis of indeterminate structures and lend technical assistance on the more difficult stress problems. In charge of structural design of XT-84, first jet plane of the company.

May 1, 1937 to Nov.12, 1940	U. S. Engineering Dept., Binghamton, N. Y. Title: Assistant Engineer Supervisor: E. S. Tippetts - Chief Engineer Duties: As assistant engineer I was responsible for the structural design of all the important hydraulic structures built by the section. The work involved initially | making feasibility studies on the Akport and Whitney Point Earth Dams by the Swedish method. Investigations on the rate of consolidation and shear stresses developed in the soft foundation were likewise conducted on this project. Of the concrete structures designed, the principal types were flip bucket, intake towers, morning glory and side channel spillways, floodwalls and large tunnels. In the
latter	stages of my study was chiefly entrusted the task of supervising the work of other engineers and establishing design procedures.
May 1, 1936 to May 1, 1937	Phoenix Engineering Corp., 2 Rector Street, New York, N. Y. Title: Cadet Engineer Supervisor: R. Stuber Duties: Conducted studies on the Santee Cooper Project (Hydraulic and hydro-electric). Also assisted in the design of power house and arch dams. In this work evolved new method of designing arch dam that reduced months of elaborate computation to matter of days.

PUBLICATIONS

1948	Article	Arch Dams with Arches of Variable Thickness	Reinforced Concrete
Feb. 1950	Article	Solution of Difficult Structural Problems by Finite Diff.	American Concrete Institute
Jan. 1951	Article	Designing for Continuity in Prestress Concrete	American Concrete Institute
Aug. 1951	Article	Analysis of Continuous Prestressed Structures	Proceedings of 1st Congress of Prestressed Concrete
Sept. 1956	Article	Hyperbolic Paraboloids and Other Shells of Double Curvature	American Society of Civil Engines Journal
Sept. 1957	Article	Ribless Concrete Cylindrical Shells	Proc. Of 2nd Symposium Con. Shell Roof Construction
Sept. 1958	Article	Practical Aspects of Ultimate Strength	American Society of Civil Engineers Journal
1960	Article	Elementary Analysis of Hyperbolic Paraboloid Shells	International Association for Shell Structures
July 1961	Article	Direct Solution of Folded Plate Concrete Roofs	International Association for Shell Structures
July 1961	Article	Design Constants for Interior Cylindrical Concrete Shells	American Concrete Institute
1963	Article	Capacity of Restrained Eccentrically Loaded Long Columns	American Concrete Institute
1963	Article	Parabolic Arches of Variable Thickness	American Society of Civil Engineers
1964	Article	Application of Shell Theory of Arch Dams	Symposium of Theory of Arch Dams
1964	Article	Design Constants for Ribless Concrete Cylindrical Shells	International Association for Shell Structures
Apr. 1966	Article	Design of Combined Frames and Shear Walls	Symposium on Tall Buildings
1966	Article	Arch Dam Layout Facilitated by Computer Usage	American Society of Civil Engineers
1968	Article	Capacity of Reinforced Rectangular Concrete Columns Subject to Biaxial Bending	American Concrete Institute
1968	Article	Floor Systems Supported by Central Core and Exterior Columns	American Concrete Institute

1969	Article	Prestressing of Flat Plates	Prestressing Concrete Institute
1969	Article	Semi-parabolic Domes of Revolution	International Association for Shell Structures

ATTACHMENT 3

ATTACHMENT 3

FORM 407-MA

PITTSBURGH TESTING LABORATORY

3901 N.W. 29th AVENUE, MIAMI, FLORIDA 33142

ORDER No. MA 8209

CLIENT'S No.

LABORATORY No. 1820(Supplement)

- 2-20-75

REPORT

ON

TESTS OF CONCRETE SAMPLES

CLIENT: Ebasco Services, Inc.
P. O. Box 1117
Jensen Beach, FL 33457

PROJECT: Florida Power & Light Co., St. Lucie Plant, Unit No. 1

DATE OF TESTS: Feb. 19, 1975, Sample Age 28 Days

The following is a report of the results of tests performed by Pittsburgh Testing Laboratory on concrete samples submitted by the client:

A. Split Tensile, ASTM C-496

Sample No.	Load, Lbs.	Split Tensile Strength, PSI
1	56,000	495
2	63,250	560

B. Flexural Strength, ASTM C-78

Sample No.	Load,Lbs.	Modulus of Rupture, PSI
1	9750	825
2	7950	670
3	7980	640

C. Modulus of Elasticity, ASTM C-469

Sample No.	Ultimate Strength,PSI	Modulus of Elasticity, PSI
1	5610	3.56×10^6
2	6030	4.18×10^6
Avg.	5820	3.87×10^6

Tech: DeRycke
cc/cc:
3 Client
1 Ft. Pierce
1 WPB
2 PG

PITTSBURGH TESTING LABORATORY

John W. Harllee

John W. Harllee, Vice Pres.

APPENDIX 3I

PERFORMANCE QUALIFICATION TEST PROGRAM

INFORMATION FOR REPRESENTATIVE BALANCE OF

PLANT CLASS 1E EQUIPMENT*

A. Switchgear - 4.16 KV

B. Motor Control Centers - 1A7, 1B7

C. Valve Operators (in containment) - Solenoid

D. Motors - Component Cooling Water Pump Motor

E. Logic Equipment - Engineered Safety Features Actuation Signals

F. Cable - 2/C #14 Shielded Twisted Pair

G. Diesel Generator Control Panel - 1A

---

* The information contained in this appendix is an abstract of equipment qualification documentation contained in FPL Quality Assurance files at the plant site. These files are available for auditing by NRC.

The test report summaries were included in response to NRC letters dated July 23, 1975 and October 10, 1975. It is not the intent to maintain the information in this appendix current. Subsequent designs will ensure that equipment utilized is equivalent or better than the existing and qualified for its application.

Pursuant to the requirements of IE Bulletin 79-01B a reevaluation of the environmental qualification of electrical equipment installed in the plant was performed. This updates the information provided in this appendix. See Section 3.11 for referencing to FPL responses to the bulletin.

In cases where the information in this section overlaps with 10CFR50.49, the Equipment Qualification List (EQ) and Documentation Packages (Doc Pacs) supersede the information provided here. Therefore, the EQ List and appropriate Doc Pacs should be consulted prior to use of the information in this section.

Information provided in this appendix is historical and shall not be updated; however, it may still be similar to the re-evaluation documentation if no changes have taken place.

A. 4.16KV SWITCHGEAR

Ebasco specification FLO-8770-284C, initially issued June 30, 1969 for the subject equipment requires the following:

1) Equipment to be built in accordance with recognized standards*

2) Equipment to be tested in accordance with recognized standards*

3) Equipment to satisfy special requirements (i.e. environmental, etc.)

* Standards listed in specification are:

IEEE
ANSI
NEMA
ASTM
ASME
NBFU
IPCEA

These standards may or may not be directly applicable to equipment purchased, ie, ASTM and ASME are applicable to oil circuit breakers, but not to oiless circuit breakers actually purchased. IPCEA standards apply only to internal wiring, IEEE standards are duplicates of ANSI and NEMA. Thus, two groups of standards are directly bearing on subject equipment's design and testing, namely ANSI and NEMA. Standards dealing with testing of medium voltage switchgear, are ANSI C37.09-1964 - "American Standard Test Procedure for AC High-Voltage Circuit Breakers" and NEMA Publication No. SG 4-1963 "High-Voltage Power Circuit Breakers". Both of those above two standards separate the tests required into 2 general categories:

a) Design tests and
b) Production tests

Category a) deals with equipment of a new design to ascertain its compliance with recognized standards. A vendors certificate, of compliance is furnished to demonstrate vendor's satisfaction of design testing to include environmental, short circuit, voltage and heat run, etc. The specified service conditions of 104F ambient (max occasional short time temperature of 49C), 35-95% relative humidity, etc, is within the design testing range.

Category b) is directly related to the production tests performed on subject equipment. Table I lists the specific tests and results achieved.

Seismic qualification data is available in Chapter 3, Appendix 3B.

Appendix 3I

TABLE I

PRODUCTION TESTS (APPLICABLE)			TESTS PERFORMED BY VENDOR
SPEC	NEMA(SG-4)	ANSI (C37.09)	
1. Hi potential Test	SG4-4.22,23	Various	1500V Secondary & Control Wiring (1 min) (1) 19KV Primary (1 min) (2)
2. Operating Tests	SG4-4.21	09-5.11	Proper operation at normal, max & min control voltage (3)
3. Operating - Interlock	-	09-5.9	
4. Correctness of Wiring	SG4-4.13	09-5.9	a) Wire check/continuity against wiring diagrams (4)
5. Operation (Devices)	-	09-5.9	b) Meters checked against laboratory standards
			c) Relays operated (4)

(Number) Refers to notes to Table I.

Appendix 3I

TABLE I (Cont'd)

NOTES:

The production tests were made to check the quality and uniformity of the vendors workmanship and materials used to manufacture the subject equipment. All production tests were made at the manufacturer's facility prior to shipment.

All production tests at the vendor's facility were either witnessed by Ebasco's Quality Compliance inspectors or certified test reports were submitted by the vendor and reviewed and accepted by Ebasco.

(1) Dielectric Tests on wiring - All secondary and control wiring were subjected to a 60 Hertz, one minute dielectric withstand test of 1500 volts. No failures were recorded, resulting in an acceptable passage of this test. Reference SG4-4.12 Item 11, SG4-4.23, NEMA standard 9-17-1953.

(2) Dielectric Tests on Major Insulation - All important insulation components, such as bushings, insulation braces, rods, etc, were subjected to a 60Hz, one minute dielectric withstand test of 19kV. No failures were recorded resulting in an acceptable passage of this test.

Reference: NEMA SG-4-4.12, Item 11.

(3) Operating Tests - Mechanical operation tests were made to check the final adjustments and to determine the ability of all breakers to operate correctly over their entire range of operating voltage without damage. Following these tests the breakers were inspected visually to verify that no parts had sustained damage.

Reference: NEMA SG-4-4.12, Item 9 and NEMA SG-4-4.21.

(4) Correctness of Wiring - Secondary wiring was checked to insure that all connections were made properly. Devices and relays, where used, were checked by actual operation where feasible. Those circuits for which operation was not feasible were continuity checked.

Reference: NEMA SG-4-4.12, Item 1 and NEMA SG-4-4.13.

B. MOTOR CONTROL CENTERS

1A7, 1B7

Ebasco Specification 210-69, entitled Motor Control Centers, was utilized for this project with an original issue date of January 22, 1971.

The specification referenced, as applicable, the requirements and test procedures established by the latest edition of the following standards:

ANSI-- American National Standards Institute
IEEE - Institute of Electrical and Electronics Engineers
ASTM - American Society of Testing Materials
NEMA - National Electrical Manufacturers Association
ASME - American Society of Mechanical Engineers
NBFU - National Board of Fire Underwriters
IPCEA - Insulated Power Cable Engineers Association

Additionally the specification references, the following standard:

EEI - Edison Electric Institute

REQUIRED TESTS

The above standards delineate design verification testing to be performed on the constituent components or completely assembled motor control centers. The intent of the referenced standards is to insure that the design test methods and procedures are capable of giving information which is pertinent, significant and reproducible.

Design testing is performed primarily on equipment of new design to verify compliance with the appropriate standards. The equipment in question utilized an established design verified by a certificate of compliance furnished by the vendor verifying that the particular design of breakers have been subjected to, and successfully passed, a test program consisting of the following:

1. A trip device test was conducted before and after the specified test sequence outlined in Table 1 of ANSI C37-50 to demonstrate the stability of the device.

2. An AC dielectric withstand test was conducted on the fully assembled breaker to demonstrate the breakers voltage withstand capability.

   Tests were conducted to demonstrate the withstand capability of the breaker between live parts and ground and across the isolating gap in the open position and in the closed position. Tests were conducted to demonstrate the withstand capability between live parts and ground and between phases.

3. A Continuous current, test was conducted to demonstrate the breaker's ability to carry 100%. rated current without exceeding the temperature conditions specified in ANSI C37-13.

4. An overload switching test was performed to demonstrate the circuit breaker's ability to carry 600%. of rated current for no less than one cycle. The number of operations are defined in ANSI C-37.16.

5. Endurance Tests were conducted in compliance with the requirements specified in C 37-16 to demonstrate the mechanical and electrical adequacy of the design.

Although the referenced standards do not mention production or acceptability testing, Purchaser's specifications require that the motor control centers be completely wired and adjusted at the factory and given standard inspection, wiring check, operation and dielectric tests. The specification further states that, the control wiring shall be factory tested as follows:

1 - Each circuit shall be given a continuity test.
2 - Each circuit shall be given an insulation resistance test with equipment connected, using a 1000 volt megger. The insulation resistance shall be not less than 25 megohms.
3 - Each starter shall be operated electrically three times.
4 - Seismic qualification data is available in Chapter 3, Appendix 3B

The specified service condition of 104°F (max occasional short time temperature of 120°F), high humidity, etc. is within the design testing testing range.

FACTORY TESTS PERFORMED

The quality compliance reports, which accompanied each motor control center, indicate that the acceptance tests required by the specification, and listed above, were satisfactorily performed prior to release for shipment.

C. VALVE OPERATORS (in Containment) - Solenoids

The qualification test program for valve operator (Solenoid) inside the containment covers design requirements, test plan, test set up, test procedures, acceptability goals and test results.

1. Design Specifications

The valve operator covered in this program was specified to meet the following codes, standards and specifications (Ref Ebasco Spec FLO-8770.110A):

.01 (IEEE) Institute of Electrical and Electronics Engineers

- 323-71
- 273
- 344
- 382

.02 (NEMA) National Electrical Manufacturers Association

.03 All valve accessories such as limit switches, positioners, solenoid shall be designed to Seismic Class I requirements.

.04 The solenoid coil shall be capable of being energized continuously without danger of overheating or failure.

.05 The solenoid coil shall be Class H (fungus proof).

.06 The solenoid coil shall be suitable for operation at 125 DC.

.07 The limit switches shall be actuated by valve stem travel and shall be supplied in waterproof housing.

.08 Each switch shall consist of two normally open and two normally closed contacts, all electrically independent. The switches, complete with mounting and actuating devices, shall be of snap acting type suitable for 125V DC operations.

.09 The solenoid coils for valves inside containment shall be totally enclosed and capable of being energized and de-energized immediately following post accident conditions.

.010 The valve operator shall be capable of withstanding an integrated dose rate of 1.0×10^7 rads during the forty-years design life of the plant.

.011 The ambient environmental conditions for all valve operator will be saturated air and steam mixture containing a maximum of 1720 ppm of boron and NaOH with a pH of 4.5 to 10.5.

2. Test Plan

The testing program covers the reviewing of vendor's test plan, test setup and test procedures. The valve operator (Solenoid) was given following tests as per Ebasco Specifications:

.01 Functional Test

Functional testing of the test item was comprised of the following series of evaluations:

a. Visual Inspection: The test item was visually examined for evidence of defects in workmanship, quality, etc.

b. Operation Test: With 100 psig applied to the inlet port of the test item, the current (I) was measured at voltage excitation of 90VRMS, 60 Hz and 125VRMS, 60 Hz.

c. Position Indication Test: The operation to the closed and open position indicators was verified for three (3) cycles minimum.

.02 Radiation Exposure Test

The test valve was exposed to a source of gamma radiation (such as Cobalt 60), for such time as to yield an equivalent exposure of 0.1 M Rad equivalent air dose per year, for forty (40) years. The entire valve was subjected to an equivalent forty (40) year radiation exposure at inside containment levels, plus the added higher dosage due to a simulated accident, (for a total of 3.3×10^7 Rads), in a single exposure at the start of test.

The radiation exposure test was conducted in a nationally recognized testing laboratory properly equipped for this purpose. Upon completion of the radiation exposure test, the test valve was subjected to the "Functional Test".

Acceptance Criteria: The valve shall meet the requirements of the "Functional Test". Discrepancies in measured data shall be brought to the attention of the cognizant Engineer for disposition.

Attachment 1 is a copy of the Radiation Certification issued by ISOMEDIX to Target Rock Corporation (valve manufacturer) on a valve operator similar to the one furnished on St Lucie Unit #1.

.03 Aging Simulation Test

A test facility as depicted in Figure 1 was prepared. Air at 100 psig was supplied to the test valve through a heater, capable of maintaining +200 ± 10°F. A hand-throttling valve located in the exit line was utilized as an adjustment to reduce the maximum flow. With the test solenoid valve open, the throttling valve was adjusted to provide a flow of approximately 10 scfm.

The test valve was installed in an environmental test chamber capable of maintaining +150°F, 55% relative humidity and ambient pressure. 115 VAC electric power through a timer switch was provided to operate the valve. The timer switch was adjustable so as to insure full valve travel in the "open" and "close" direction as monitored by the counters. The counters were connected to each position switch, and thus indicated the number of full cycles completed.

The 27 ft^3 Conrad Environmental Chamber was utilized for the aging test and for portions of the accident simulation test. This chamber is equipped with continuous recording and programming capability and can also perform simultaneous temperature-altitude and humidity testing.

Actual accident simulation was accomplished, by injecting steam into a stainless steel pressure vessel which incorporated remotely actuated pressure control valving.

.04 Temperature-Humidity and Wearout Test

100 psig air was supplied to the test valve (heated to +200 ± 10°F), and the chamber temperature was raised to +150 ± 10°F with a relative humidity of 55± 5%. Cycling of the valve was then started.

The valve was cycled by supplying 115 VAC power to the solenoid coil through the timer switch. The timer was adjusted to insure completeness of each cycle by comparison of the counters for similarity of count.

The cycling period was continued until 1000 cycles had been completed, at which time the cycling of the valve was stopped. The internal chamber conditions were maintained until 240 hours had been completed. Chamber conditions were recorded by the continuous chart method.

At the completion of this test, the test item was subjected to the "Functional Test", described earlier.

.05 Accident Simulation Test

A test facility similar to that utilized for the Temperature-Humidity and Wearout Test was prepared with the exception that the environmental chamber was fabricated utilizing a stainless steel pressure vessel capable of withstanding 100 psig internal pressure. Steam and water inlet and drain ports were provided as well as a means of maintaining the internal temperature at +350°F. No air connection was made to the test valve. The wiring from the junction boxes was teflon and epoxeyed through feed-through fittings in the chamber wall. The junction boxes were epoxeyed.

Steam, air, and de-mineralized water were injected into the test chamber, raising the pressure to 70 psig, the relative humidity to 100%, and the temperature to +300 °F in as close to ten (10) seconds as possible, and these conditions were allowed to stabilize. The valve was exercised OPEN and CLOSED, and operation was verified. The temperature was then increased to +340°F within five (5) minutes.

These conditions were maintained for three (3) hours. The valve was periodically exercised for a total of ten (10) times. The environmental conditions were then reduced to ambient over the following two (2) hour period.

Steps a and b (above) were then repeated after which the chamber temperature was adjusted to +250°F and the pressure to 25 psig. These conditions were maintained for a period of four (4) hours. The valve was exercised approximately once each hour. The pressure was then reduced to ambient and the temperature was maintained at +200°F for a period of thirty (30) days. The valve was exercised once each working day.

At the conclusion of the test, the valve was subjected to the "Functional Test" described earlier.

.06 Seismic Qualification Test

The valves used in the qualification test program were qualified by seismic calculations. The seismic calculations were reviewed in compliance with Ebasco specifications for solenoid valves. A similar valve operator was also tested in a seismic testing facility to prove the operability of the solenoid under the seismic conditions. The following testing plan and procedure was followed:

The seismic simulation facility provides a low frequency, high displacement excitation utilizing hydraulic actuators. This facility can be utilized in the uni-axial mode as well as the bi-axial mode. The basic hydraulic system can provide up to 200 GPM to systems employing Moog 72-102 or 72-103 Servo-valves. Actuator displacements up to 10 inches at 60 ips can deliver up to 25,000 force-pounds at 2500 psi. The frequency range of the system is 0.1 to 100 Hz. nominal with availability to 500 Hz. The uni-axial system normally employs three (3) 60 gpm Servo-valves with provisions for one (1) additional valve. The bi-axial system employs two (2) or three (3) 60 gpm valves and two (2) 40 gpm valves.

The system is capable of performing: Continuous Sine, Sine Beat, Decaying Sine and/or Random Excitation.

The uni-axial test facility (see Figure II) was utilized for performance of the testing. The fixture was designed so as to be free of resonance from 0.1 to 33 Hz. The counters and actuating switch were provided to permit actuation and monitoring during and after each period of vibration.

FOR SEISMIC QUALIFICATION TESTS

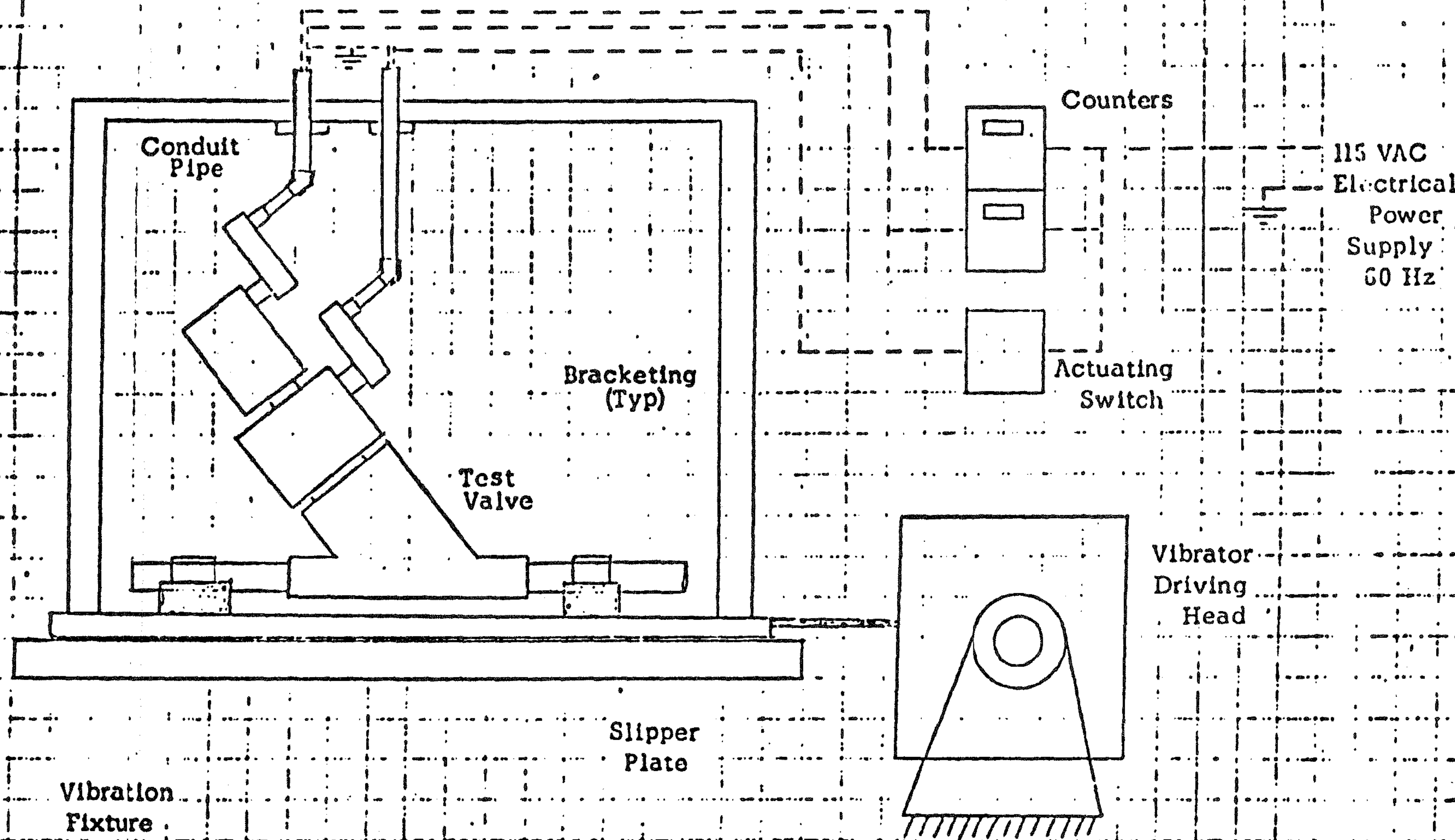


FIGURE 3I-I

TRC #1383

TEST SCHEMATIC FOR AGING TEST

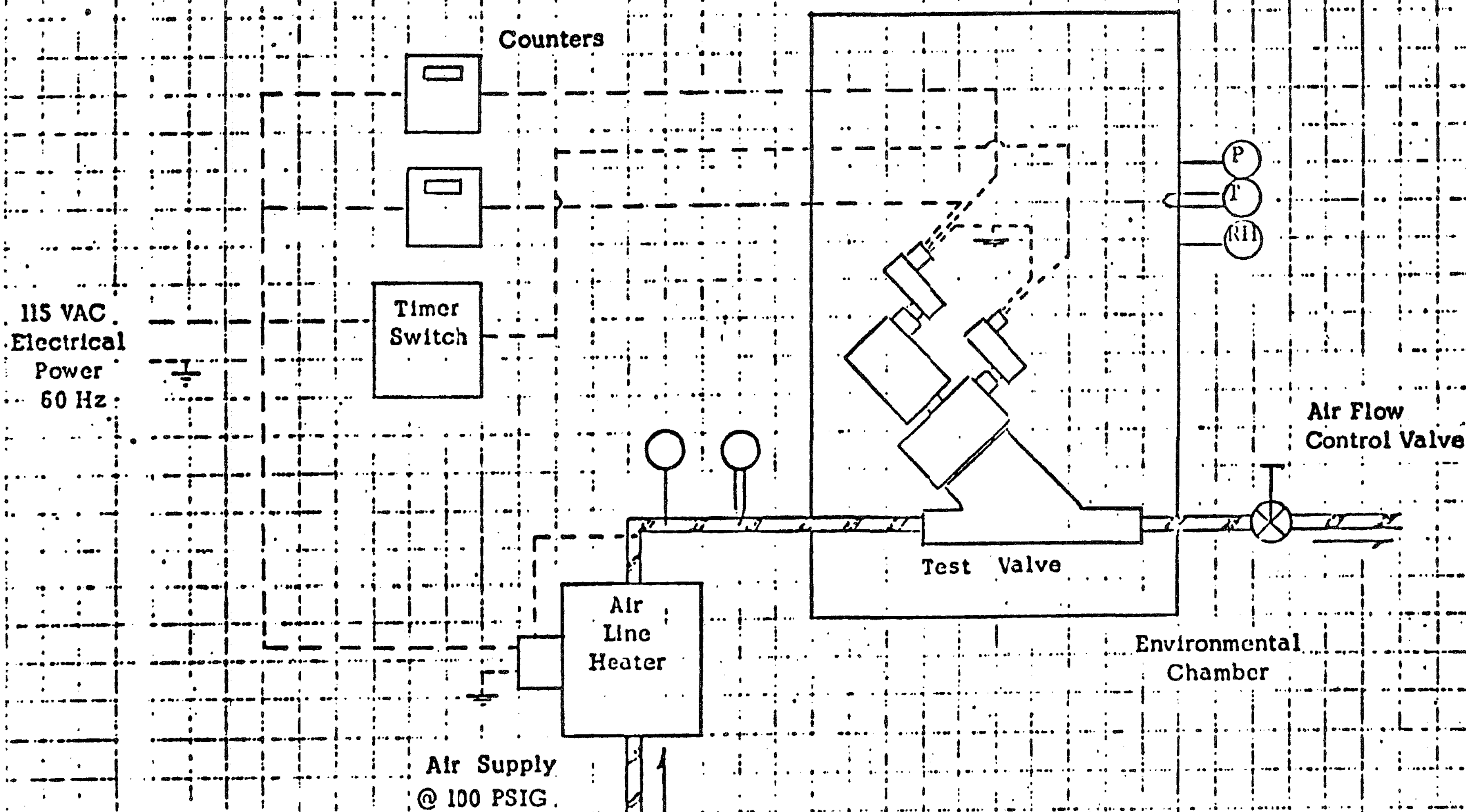


FIGURE 3I-II

TRC # 1383

The valve was installed in the vibration test fixture and actuated to insure proper counter function.

An exploratory vibration test was conducted in order to determine the natural frequencies of the test item and/or its various components. The scan was conducted at 0.2g's or to the limits of the system over the frequency range of 0.1 to 33 Hz. The resonant frequencies, (if any) were recorded.

The unit was subjected to 3 g's of vibration (or to the limit of the system), at each of the noted resonant frequencies for a period of ten (10) seconds. An actuation was performed during and after each resonant period.

The unit was also subjected to 4.5 g's of vibration (or to the limit of the system) at each of the resonant frequencies for an additional period of ten (10) seconds. An actuation was performed during and after each resonant period.

This procedure was performed along each of three (3) mutually perpendicular axes of the unit. After each test, the valve was actuated to insure that no deterioration has taken place as a result to this exposure.

At the conclusion of the seismic qualification test, the test unit was subjected to the "Functional Test" described earlier.

3. Acceptability Goals & Test Results

The results of all tests were reviewed to check compliance with the design specification. The certificate of compliance, stating that the valve operator meets the design requirements specified in Ebasco Specification FLO 8770.110A, was obtained from the valve manufacturer and is appended as Attachment 2.

IRRADIATION CERTIFICATE
ATTACHMENT 1
ISOMEDIX

May 20, 1974

Target Rock Corporation
1966 E. Broadhollow Road
E. Farmingdale, New York 11735
Attention: Mr. Arthur Summertano

Gentlemen:

Attached is our certification for the irradiation of a Standard Production Valve, SN X-1.

From a radiation standpoint, the valve contains no added or induced radioactivity, and from a radiation standpoint is completely safe to handle.

Very truly yours,

George R. Dietz

George R. Dietz
Manager, Radiation Services

Isomedix Inc.. 25 Eastmans Road, Parsippany, New Jersey Telephone (201) 887-4700
Mailing Address: Post Office Box 177, Parsippany, New Jersey 07054

Isomedix Limited. Benoit Street, Mont. St. Hilaire, Quebec, Canada Telephone (514)467-1211
Mailing Address: Post Office Box 7, Benoit, Quebec, Canada

ATTACHMENT 1 (Cont'd)
ISOMEDIX
RADIATION CERTIFICATION 5-20-74

Part No: Target Rock Corp
73 E-001 Standard Production Valve, SN X-1

Dose Rate: 0.5 Mrad/hr.

Total Dose: 33 Mrad, air equiv. Maximum overdose factor: 1.2

Date Radiation Completed: 5-20-74

Source: Cobalt-60

Conditions: Irradiation performed in air at ambient temperature (70°F) and slight negative pressure (-1/2" water).

Max. Temp. of Sample During Irradiation: 110°F

Dosimetry System: Dosimetry was performed using a Victoreen Model 555 Integrating Dose Rate Meter and Probe. The unit was calibrated on January 15, 1974 by the Victoreen Instrument Company, using Cobalt-60 and Cesium-137 sources whose calibrations are traceable to the U. S. National Bureau of Standards. A copy of the calibration certificates is available.

Other: Samples were rotated and turned during exposure to achieve a more uniform dose distribution.

Post-Irradiation Defects Observed:

None
Sample contains no added or induced radioactivity.

This is to certify that the subject product was radiation processed in the aforementioned manner.

George R. Dietz
George R. Dietz
Manager, Radiation Services

GRD:mg

Isomedix Inc.. 25 Eastmans Road, Parsippany, New Jersey Telephone (201) 887-4700
Mailing Address: Post Office Box 177, Parsippany, New Jersey 07054

Isomedix Limited. Benoit Street, Mont. St. Hilaire, Quebec, Canada Telephone (514)467-1211
Mailing Address: Post Office Box 7, Benoit, Quebec, Canada

ATTACHMENT 2

Target Rock Corporation

CERTIFICATE OF COMPLIANCE

Reference: EBASCO ORDER

Purchase Order Number: NY422362 F 0 R

FLORIDA POWER & LIGHT COMPANY

T.R.C. Project 74Q

TAG NUMBERS:	I-SE-02-1	I-SE-02-3
	I-SE-02-2	I-SE-02-4

We certify that the parts and assemblies noted herein meet all the requirements of the purchase order drawings and specifications. Further we certify that the detailed fabrication inspection and test records quoted in the order and specifications are traceable and on file. Final cleaning and drying conforms to Target Rock Corporation Procedure 890, Rev. 1.

3/14/75

G. Abruzzo, Date

Manager of Quality

D. MOTORS - Component Cooling Water Pump Motor

Ebasco Specification 8-69 "Motors For Station Auxiliary Service" bearing Purchaser's Identification Number FLO-8770.289 initially issued July 15, 1969 requires compliace with the following standards - USASI (ANSI), IEEE, ASME, NEMA and AFBMA. The standard directly applicable to production testing is IEEE standard No. 112A (September 1964) (Revision of AIEE No. 500 1954) - "Test Procedure For Polyphase Induction Motors And Generators". The scope of this standard is as follows: "This test procedure covers instructions for conducting and reporting the more generally applicable and acceptable tests to determine the performance characteristics of polyphase induction motors and generators. It is not intended that the procedure cover all possible tests or tests of a research nature. The procedure shall not be interpreted as requiring the making of any or all of the tests described herein in any given transaction".

The Ebasco specification itself requires the following tests:

a) Factory routine test or initial short commercial test

b) Qualification of sealed insulation system unless previously approved by Purchaser.

The factory routine test, as per IEEE 112A, includes: "....measurement of speed and current input at no load, current input with locked rotor, and dielectric tests ... "

These routine tests were performed with the following results for the component cooling water pump motor, item 5, of General Electric Motor Order 300-94977, NY-422208.

Note: CCW 1A Motor replaced with Homewood Energy Services Motor.

Serial Nos.	8380428	8380429	8380430	Vendors Acceptance Criteria
Impedance Amps. (FLA.)	66	65.52	64.8	±10% of
Excitation Amps	16.08	16.56	17.28	Calculated
Running Light Watts	8448	9360	8160	Values
Station Winding Cold Resistance (OHMS)	0.9760	0.9650	0.9710	+ 5%
Motor End Play	30/64"	37/64"	38/64"	Within coupling limits
Hipot (9000V/1min)	Passed	Passed	Passed	--
Speed (rpm)	1800	1800	1800	--
Vibration	<0.4 mils	<0.4 mils	<0.4 mils	0.0025 (Maximum)

Stator Winding IR	1000MΩ	1000MΩ	1000MΩ	______
Heaters				
Hipot (1000V/ 1 min)	Passed	Passed	Passed	______
IR	1000MΩ	1000MΩ	1000MΩ	______
Resistance	16.2Ω	16.8Ω	16.6Ω	______
Noise Level	Passed	Passed	Passed	90DBA

The above tests were conducted in accordance with IEEE 112A with the following setups.

1 - Speed and Current Input at No Load

These tests are in accordance with IEEE No. 112A, Paragraphs 4.6 and 4.9. The test with no load is made by running the machine at rated voltage and frequency without connected load. To insure that the correct value of friction loss in obtained, the machines are operated until the input becomes constant. The current is read in each line. The speed current characteristic is the relation between current and speed. The speed torque and speed current tests may be made with a dynamometer or a calibrated machine as the load. Measurements of current voltage and speed are made. The torque output is obtained directly from dynamometer readings or indirectly (by calculation) from calibrated machine measurements.

The speed-torque and speed-current tests whall be made at rated voltage or as near to it as is practical.

2 - Current Input with Locked Rotor

This test is in accordance with IEEE No. 112A dated Spetember 1964 Paragraphs 4.8. This test may be taken either as a check of quality or to determine performance. When possible, readings were taken at rated voltage and frequency, since the current is not directly proportional to the voltage because of changes in reactance caused by saturation of the leakage paths.

3 - Dielectric Tests

Dielectric tests are in accordance with IEEE No. 112A, paragraph 6.2. Measurement of dielectric test voltage are discussed in American Standard Measurement of Voltage in Dielectric Tests, C68, 1-1953 (IEEE No. 4). The voltmeter method of measurement is commonly used.

The dielectric test voltage was successively applied between each electric circuit and the frame with the windings not under test and the other metal parts connected to the frame. Interconnected polyphase windings (2 phase 4 wire included) are considered as one circuit.

No loads were left unconnected during the test as this could cause an extremely severe strain at some point in the winding. in making the test, the voltage was increased to full value as rapidly as possible while still maintaining an accurate meter reading, and the full voltage maintained for one minute. It was then reduced to one-quarter value or less in not more than 15 seconds.

The following environmental conditions were specified and attested to by the Vendor:

Location - outdoor

Altitude - sea level

Ambient temperature range - (30°F to 120°F)

Site Conditions - subject to torrential rains, and salt-laden atmosphere.

E. LOGIC EQUIPMENT - Engineered Safety Features Activation Signals

1. Equipment Design Specification Requirements

The logic panels for the initiation of the engineered safeguards were designed in strict conformity with the IEEE No. 279 "Criteria for Nuclear Power Plant Protection System" and contain control and instrumentation used to generate the following actuation signals:

a - Safety Injection Actuation Signal (SIAS)
b - Recirculation Actuation Signal (RAS)
c - Containment Spray Actuation Signal (CSAS)
d - Containment Isolation Signal (CIS)
e - Main Steam Isolation Signal (MSIS)

These actuation signal logics consist of solidstate circuits using standard reliable hardware of modular design for easy servicing and/or replacement.

The logic panels are designed in accordance with St. Lucie FSAR, particularly Sections 3.10, 3.11, 6.2, 6.3, 7.1, 7.3, and 17.

The instrumentation and controls which actuate and control the engineered safeguards systems were designed on the following bases:

a - Redundant instruments are provided for initiating safeguards systems action. No single component failure presents safety action.

b - Four sensors are used for each of the critical parameters.

c - The operation of any two of four sensors must initiate the appropriate engineered safeguards action. (Two out of three when one sensor is disconnected for maintenance).

d - Redundant circuit wiring run in separate wiring raceways and power supplies are fed from separate high quality instrument buses, consistent with the principle of maintaining independence of channels.

e - A loss of instrument power to the measurement channels and logic systems initiates engineered safeguards actuation signals. All measurement channels and logic matrices assume the nonconducting state when initiation of safeguards occurs. (Except as noted)

f - Each circuit component in all the systems is periodically tested and visually verified of its proper functioning or its malfunctioning. Malfunctions are commonly annunciated.

* - Except where noted otherwise

g - All equipment, including panels, components and cables associated with the engineering safeguards systems are marked with colored markers or nameplates in order to facilitate its identification. The following colors are used:

Measurement Channel A - Red
Measurement Channel B - Yellow
Measurement Channel C - Green
Measurement Channel D - Blue
Safety Channel A - Orange
Safety Channel B - Purple
Safety Channel A - Pink

h - Electrical circuit separation is provided between engineered safeguards systems and annunciations, data loggers and computers.

i - Environmental Conditions: The equipment is located in an air-conditioned control room. Equipment performs as specified when subjected to the following environmental conditions.

Temperature, Control Room: Expected normal - 72 F
Operating Range - 40 F to 130 F
Ambient Humidity Range: 40% to 95% max relative humidity
Line Voltage - 115 v a-c ±10%
Line Frequency - 60 H_z ± 5%

2. <u>Test Plan</u>

Preproduction modules are subjected to the following tests:

a - Operating (including range adjustment)
b - Temperature
c - Power Consumption
d - Supply Voltage

Testing of current production modules which have undergone above test is not required provided test results are made available to Florida Power & Light Co.

The following tests are performed to assure conformance with requirements of this specification:

a - Visual Examination (all units)
b - Operating (all units)
c - Temperature (on a sample)
d - Dielectric Strength (on a sample)
e - Insulation Resistance (all units)
f - Noise Susceptibility (on a sample)

Florida Power & Light reserves the right to review and evaluate test data on previously produced designs for assurance that the equipment will conform with the requirements of this specification.

The system testing demonstrates the compatibility between assemblies and the ability of the equipment to be operated as a system. Simulated input signals and monitoring for outputs are utilized in this testing.

System testing includes:

a - Visual examination
b - Operating test

1 - Operating test may be used to confirm correct system wiring.
2 - Point-to-point wiring check of all wiring not included in the operating test. (Example: A board mounted handswitch with wiring for external use by others.)

c - Dielectric strength of wiring
d - Insulation resistance of wiring
e - Supply voltage
f - Power consumption
g - Module interchangeability
h - Noise susceptibility

In addition, the equipment is operated for four weeks as a "burn-in" test to demonstrate its stability to determine any design oversights and to provide time for the development of any "infant mortality" component failures. System testing time may be counted as part of the "burn-in" test time.

The "burn-in" includes two weeks accelerated life testing. This test involves operating at the undervoltage and overvoltage extremes with periodic (minimum one per day) 15 minute de-energization of the equipment.

3. Test Set Up, Procedures and Acceptability Requirements

The following test descriptions define the requirements of the tests specified in the proceeding paragraphs.

a - Visual examination of:

Workmanship, assembly and fit
Materials, parts and finishes
Treatment for prevention of corrosion
Safety requirements
Marking

b - Operating: The equipment shall be energized and subjected to an operating test to ensure proper functioning including all controls and adjustments. It shall be demonstrated that the alignment procedures to be incorporated in the technical manual are adequate for aligning equipment. In devising the test procedure, reference shall be made to the component specification for the operating limits and accuracies and other operational requirements.

Tests shall demonstrate that the equipment meets the functional requirements of this specification, particularly in regard to obtaining outputs with the correct combination of inputs.

Demonstration that sufficient isolation is provided so that interaction between redundant channels or between protective channels and measurement circuits does not exist.

Demonstration that sufficient interlocks exist so that normal and abnormal test procedures do not produce dangerous conditions. For example, testing in an unusual sequence.

Demonstration that interlocks do not prevent normal operation of the plant.

c - Temperature: Equipment shall be tested at the minimu, 77 F ± 10 F and maximum equipment ambient temperatures based on specified control room temperature and allowing for the enclosure temperature rise. Equipment shall meet the component temperature specifications. (Control room temperature)

d - Power Consumption: Equipment shall be checked to determine power required from the supply line or power supply. Power or current from each supply shall be recorded.

e - Dielectric Strength: Complete assemblies shall be subjected to a dielectric strength test. The test voltage specified below shall be applied for a period of not less than 1 minute except that for equipment requiring a test voltage of 2,500 volts or less, a test voltage 20 percent higher than that specified may be applied for a period of not less than 1 second. Before testing an assembly having circuits grounded to chassis, the ground to chassis shall be disconnected. Shielding shall not be disconnected from ground. The test voltage shall be applied between each electrically isolated circuit and ground. After completion of this test, there shall be no evidence of punctured insulation or arcing.

The test voltage shall approximate a true sine wave and shall have a frequency not less than the rated frequency of the circuit under test. The magnitude of the test voltage shall be determined as follows:

The dielectric test voltage shall be 500 volts AC.

f - Insulation Resistance (cold): Completed assemblies shall be subjected to an insulation resistance test. Prior to application of the test voltage, the grounds for all grounded circuits shall be disconnected from ground. The test voltage shall be applied between all electrically isolated circuits and between each circuit and ground. Criteria for establishing the boundaries of an electrically isolated circuit shall be as specified for the dielectric strength test. When testing between the circuits and ground, all circuits may be tied together so that only one test voltage need be applied, providing that the insulation resistance when tested in this manner meets the minimum value specified in the individual component specification. The time of test voltage application shall be not less than 60 seconds. The test shall be conducted with the equipment cold at any convenient room temperature. The temperature of the circuits at the time of the test shall be measured and recorded.

g - Supply Voltage: The equipment shall be operated at the extremes of supply voltage specified for the system. Line voltage operated equipment shall be tested at the extremes of line voltage specified in Section 1. Equipment operating from DC power supplies will be tested with voltage variations equal to or greater than those anticipated or measured with the combined effects of line and load regulation.

h - Module Interchangeability: Evidence of module and assembly interchangeability shall be provided. Modules and assemblies refer to NIB modules or other plug-in or readily changeable assemblies that will be stocked as spare parts.

i - Noise Susceptibility: Electrical noise tests shall be determined by the designer. Tests shall endeavor to ensure the equipment will operate satisfactorily in the presence of radiated and conducted electrical noise.

Certified performance data (from actual tests) shall include conformance with IEEE 279 "Nuclear Power Plant Protection Systems" dated June 3, 1971. Compliance with Paragraph 4.2 of IEEE 279 shall be demonstrated by performing a fail safe and fault analysis of all components in the system.

Consolidated controls shall also perform a reliability analysis for all equipment supplied with this specification. This analysis shall describe the system reliability in terms of "Mean Time Between Failure (MTBF)" and recommend test intervals to maintain the reliability at various levels.

It is the intent of this document to demonstrate that the equipment meets the IEEE Standards for Nuclear Power Plant Protection Systems and NRC "General Design Criteria for Nuclear Power Plant Construction Permits."

Further compliance with the following is required:

a - IEEE 279 "Nuclear Power Plant Protection Systems"
b - IEEE 308 "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations"
c - AEC Publication TID 7024 "Nuclear Reactors & Earthquakes"

Engineered safeguard logic panel seismic qualification is documented in FSAR Chapter 3, Appendix 3B.

4. Certification and acceptability of this equipment is documented in Attachments 1 and 2, appended.

Attachment 1

CERTIFICATION

Engineered Safeguards Logic Panels
St. Lucie Plant Unit No. 1

The Engineered Safeguards Logic Panels are manufactured and tested in accordance with approved Drawings and Test Procedures in accordance with the Purchase Order requirements and specifications of Purchase Order NY 422292 and Specification FLO-8770.145.

The quality assurance package contained herein verifies that the Engineered Safeguards Logic Panels, Part No. 9N17 are in accordance with these purchase specifications, drawing revisions, and test procedures.

Certified by:

____________________ Date 10/22/73

G. L. Schoenbaum,
Project Engineer

____________________ Date 10/22/73

L. Goodsell,
Quality Assurance Manager

ATTACHMENT 2

CONSOLIDATED CONTROLS CORPORATION
15 DURANT AVENUE, BETHEL, CONNECTICUT 06801 203 743-6721
TWX 710-456-0445 TELEX 969615

A SUBSIDIARY OF CONDEC CORPORATION

February 14, 1974

Ebasco Services Inc.
2 Rector Street Room 800
New York, New York 10006

Attention: Mr. Vito Oniunas

Subject: Qualification of Engineered Safeguard
Logic Panels
St. Lucie Unit # 1

Gentlemen:

Enclosed is the certification of the humidity and seismic qualification items per our telecon of 2/12/74. We are pleased to be of assistance at this time.

Very truly yours,

CONSOLIDATED CONTROLS CORPORATION

G. L. Schoenbaum
Product Manager, Systems

/Lmp

Enclosure: Certification
Engineered Safeguard Logic Panels
St. Lucie Nuclear Power Station
Unit # 1
Dated February 14, 1974

ATTACHMENT 2 (Cont'd)
CERTIFICATION
Engineered Safeguard Logic Panels
St. Lucie Nuclear Power Station
Unit # 1
February 14, 1974

I Engineered Safeguard Logic modules 6N81 through 6N87 similar to 6N88, 6N89, 6N90, 6N91 and 6N92 were temperature qualified to 130 degrees Fahrenheit and 94% RH. The modules met all specification during the seven day test. Consolidated Controls Corporation Engineering Report # 803 dated 2/21/73 incorporates the procedure and data of this qualification.

II Engineered Safeguard Logic actuation relays S. H. Couch type 4CP36-AF were tested both in the energized and de-energized state during seismic qualification of the St. Lucie I equipment. Proper operations of the relays were verified prior to and subsequent to the application of the seismic forces. This data is contained in Consolidated Controls Corporation Engineering Report # 824 dated 12/5/73.

In addition to the above qualification testing, Consolidated Controls Corporation has previously qualified actuation relays built by S. H. Couch Company. Consolidated Controls Corporation Engineering Report # 771, Confidential Restricted Data dated 12/5/69, documents the operation of a S. H. Couch 4AP37-AF relay during a MIL-STD-167 vibration test. After three hours of vibration, two relays were switched due to the application

CERTIFICATION
Page Two

of trip imput driving signals. The equipment tested was Reactor-Protective Equipment being provided to the U. S. Navy. The 4AP37-AF relay is of the same construction as the 4CP36-AF relay being used in the St. Lucie I equipment. The only differences are the "A-P" which changes contact current capacity and the "37-36" which changes the coil impedence. The relay operated properly during the vibration and met all specified performance requirements.

CONSOLIDATED CONTROLS CORPORATION

G. L. Schoenbaum
Product Manager, Systems

/lmp

F. CABLE - 2/C #14 Shielded Twisted Pair

The item selected as an example is from Purchase Order NY-422273, General Cable Corporation, Bill of Material Item D4-7: one-pair (2 conductor) #14 AWG, 7 strands, soft drawn coated copper, 30 mils XLPE unfilled insulation (90 C rating), 15 mils color coated PVC jacket over insulation, cabled with Dupont Hyten Type 102 nylon fillers and overall mylar tape, #16, 7 stranded bare soft drawn copper drain wire, copper mylar shielding tape and 45 mils PVC overall jacket.

The purchase order specification Ebasco Specification FLO-8770-292A initially issued March 5, 1971, stated "The requirements and test procedures established by the latest ANSI, IEEE, NEMA, ASTM, ASME, NBFU, UL, AEIC and IPCEA standards, whenever applicable, are in force as minimum requirements. Also the specification called for the following additional test requirements to be met:

1) Single completed conductors to meet UL44 horizontal flame test

2) Prior to cabling, the individual insulated conductors shall be given a 2500 volt a-c spark test by grounding the conductors and appling a voltage by use of flexible metal fingers around the insulated conductors as the conductors are cabled.

3) The completed cable shall be given:

   a) Dielectric Tests - 1000 volts a-c for 5 minutes, conductor to conductor, and conductor to shield.

   b) Sheath shall be subjected to an a-c spark test of 3000 volts minimum.

   c) Insulation resistance measurements per IPCEA S-19-81, Paragraph 6.23.

4) Jacket material shall have a maximum gravimetric water absorption of 40 milligrams per square inch after immersion in water at 70°C for seven days.

5) The jacketed cable shall pass the IPCEA vertical flame test.

6) The completed cable shall withstand a total radiation dose of 5.0×10^5 rads.

7) Cable shall be suitable for operation for a period of up to 15 minutes post LOCA. (FSAR Section 3 Environmental Category 1B).

The basic document for factory testing is IPCEA S-66-524 entitled "Cross-Linked-Thermosetting-Polyethylene-Insulated Wire And Cable For The Transmission And Distribution Of Electrical Energy". Part 6 of this document covers "Testing And Test Methods". Vendor's certificate of compliance states that this has been complied with. Results of these tests are part of the QA project documentation files.

The results of the additional testing required by the specification is as follows. (Numbers refer to above-mentioned additional test requirements)

1. The completed single conductor passed the UL44 horizontal flame test.

2. The spark test was run as part of the cabling process.

ELECTRICAL CABLE (Cont'd)

3. a) and b) a 4.0KV a-c Voltage was applied for 5 minutes with no failure observed
   c) Insulation resistance measurements per IPCEA S-19-81, Paragraph 6.23 were 26400 megohms per thousand feet. Minimum acceptable reading is 3950 megohms per thousand feet.

4. The water absorption (gravimetric method) was observed to be 7.3 milligrams per square inch.

5. The jacket cable material successfully passed the IPCEA vertical flame test.

6&7. A sample of this cable was subjected to a radiation dose of $5x10^5$ rads and then subjected to post-LOCA condition of steam and chemical sprays by the Franklin Institute.

The environmental conditions were as follows:

a) From ambient to 50 PSIG and 286 $^\circ$ F within 5 seconds.

Time (Hrs)	Pressure(PSIG)	Temp (^{O}F)
3	49	269
24	6	161
71	5.5	161
121	5.5	160
181	5.5	162
196	0*	80*

* Prior to removal of cable from chamber.

The chemical spray, consisting of 2000 ppm of boron as boric acid buffered with NaOH to a pH of 9.0 was turned on 30 seconds after the test started and remained on throughout the entire test period. The spray was maintained at a rate of 0.6 gpm.

The cables were electrically energized throughout the steam chemical-spray exposure at 300 volts a-c between conductors. The power was turned off approximately one hour prior to each set of insulation resistance measurements and tuned back on immediately after the measurements were made. The minimum insulation resistance reading was $3.2x10^8$ ohms.

The conclusions of the Franklin Institute were as follows: "Nine samples were submitted for test. Eight samples were exposed to gamma radiation for a total accumulated (equivalent air) dose of 0.5 megarad. One cable was retained as a control element. Four of the eight cables that were exposed to gamma radiation were further subjected to a steam/chemical spray environment that lasted seven and one-half days. The cables were electrically energized to 300 Vac with no current, during the radiation and steam/chemical-spray exposures. Measurements of insulation resistance were made periodically throughout the test program. No electrical difficulties were encountered and the cables were considered to have performed satisfactory throughout the test program".

General Cable Corporation further stated: "The results of physical properties and voltage tests after irradiation, steam and chemical exposure of the completed cable indicate retention of all essential properties for satisfactory electrical performance".

G. DIESEL GENERATOR CONTROL PANEL - 1A

Purchase Order Specifications FLO-8770-305 and FLO-8770-290 initially issued January 30, 1970 and December 15, 1968 respectively, required the use of Specification FLO-8770-290 for the construction of the diesel generator control panel. The production test requirements for the control panel were a) a continuity check and b) a 1000 volt meggar check of all non-electronic circuits with a minimum reading of 25 megohms.

The purpose of the continuity check (a) above, was to insure that the wiring was in accordance with the vendor's drawings. The procedure is to check continuity between various terminals to insure that the circuit is continuous as indicated by the applicable drawings. The test results of the diesel control panel indicated that the control panel was wired in accordance with the applicable drawings and tested accordingly.

The purpose of the 1000 volt meggar check is to insure that the integrity of the wiring is maintained. The procedure is to apply the voltage to each terminal point (following the disconnection of all electronic equipment which can be damaged by 1000 volts) and record the meggar reading. Any reading of less than 25 megohms requires that the circuit be checked and a repair made which will result in the proper reading. The result of this test was a minimum reading of 40 megohms.

In addition to the control panel check, the purchase order specification required various functional checks of equipment on the diesel generator. The results of these tests were submitted to the staff as a proprietary document. A review of this test indicates that the control panel operated as specified for all operating conditions including a) the 300 start test and b) normal performance tests.

All tests performed by the diesel generator control board vendor were witnessed by Ebasco Quality Compliance Inspectors with the results as indicated in the above-mentioned report.

ENVIRONMENTAL REQUIREMENTS

The purchase order specification listed the following service conditions:

1) Location -.Indoor

2) Altitude - Sea Level

3) Ambient Temperature - 40°C (104°F)

4) Location of Air Intake - Outdoor

5) Location of Exhaust – outdoor

6) Site Conditions - The plant site is subject to torrential rains, hurricane winds, floods and a salt-laden atmosphere.

The control panel was fabricated Nema 1 with thermostatically controlled space heaters to preclude condensation. The control panel utilized industrial components which will operate in a 40°C ambient condition.

ENVIRONMENTAL REQUIREMENTS (Cont'd)

Vendors certificate of compliance states that all requirement of the purchase order specification have been complied with.

SEISMIC TESTING

Due to the difficulty in supporting the integrity of electrical continuity where required to maintain the diesel generator in an operative mode by calculation, actual simulated tests have been performed. The electrical cabinet was excited bi-directionally from 1 through 35 Hertz and at acceleration forces up to 0.35 g.

TEST REQUIREMENTS

a) Resonant search

A sine sweep from 1 Hertz to 33 Hertz at a logarithmic rate of 1 octave per minute and at a level of 0.20g horizontally and 0.1g vertically was preformed.

b) Full level tests

Sine dwells at the six most significant resonant frequencies was performed (7,8,11,15,23,33Hz). Force levels in gravity units were 0.3 horizontal and 0.2 vertical. The duration of each dwell test was 30 seconds.

c) Circuit monitoring

Electrical circuits were monitored to ascertain electrical continuity, current/voltage levels, spurious operation, contact chatter, etc., before, during and after the seismic excitation.

d) Insulation resistance

Insulation resistance measurements were performed before and after seismic tests on two electrical circuits.

TEST PROCEDURES AND RESULTS

a) Resonant search

During the sine sweep from 1 to 33 Hz chatter occurred on the lockout relay signal from 5 to 6 Hz. A CFD relay was temporarily disconnected and the test resumed. No further chatter occurred. The rear door was reinforced with 1"x1" angle iron in the area of the CFD relay. The sine sweep was performed again with lockout relay chatter at 15 Hz. A short dwell test was then performed at 15 Hz with no relay chatter. Again a sine sweep was performed with lockout relay chatter at 15 Hz.

The rear door was bolted at the top and bottom to simulate-door latches. The bolts allowed the rear door to be fastened at four points. The sine sweep was performed again with no chatter occurring. Six responsive frequencies were then selected for full level dwell testing. (7,8,11,15, 23 and 33 Hz).

<u>TEST PROCEDURES AND RESULTS</u> (Cont'd)

b) Full level testing

Steady state sine dwell testing was performed at frequencies of 7, 8, 11, 15, 23 and 33 Hz. The input during these dwell tests were:

Horizontal Input 0.3g peak

Vertical Input 0.2g peak

In addition, at 15 Hz the inputs were raised to .37g horizontal and .25g vertical to insure the integrity of the lockout relay.

All monitored circuits performed as required during the sine dwell test.

c) Circuit monitoring

Following the reinforcement and bolting of the rear door, no further relay chatter occurred.

d) Insulation resistance

The insulation resistance measurements were taken from a terminal point to ground (readings in megohms):

Circuit Tested	Before Test	After Test
100	20	17.5
102	21	17.0

<u>CONCLUSION</u>

To insure the reliability of the structure, addition of formed gussets and/or channels were added to each compartment with emphasis on directing the stress toward the base of the cabinet. This elevated the natural frequency a greater amount over the floor spectra.

TEST PROCEDURES AND RESULTS (Cont'd)

b) Full level testing

Steady state sine dwell testing was performed at frequencies of 7,8,11, 15,23 and 33 Hz. The input during these dwell tests were:

Horizontal Input	0.3g peak
Vertical Input	0.2g peak

In addition, at 15 Hz the inputs were raised to .37g horizontal and .25g vertical to insure the integrity of the lockout relay.

All monitored circuits performed as required during the sine dwell test.

c) Circuit monitoring

Following the reinforcement and bolting of the rear door no further relay chatter occurred.

d) Insulation resistance

The insulation resistance measurements were taken from a terminal point to ground (readings in megohms):

Circuit Tested	Before Test	After Test
100	20	17.5
102	21	17.0

CONCLUSION

To insure the reliability of the structure addition formed gussets and/or channels were added to each compartment with emphasis on directing the stress toward the base of the cabinit. This elevated the natural frequency a greater amount over the floor spectra.

TEST PROCEDURES AND RESULTS (Cont'd)

b) Full level testing

Steady state sine dwell testing was performed at frequencies of 7,8,11, 15,23 and 33 Hz. The input during these dwell tests were:

Horizontal Input	0.3g peak
Vertical Input	0.2g peak

In addition, at 15 Hz the inputs were raised to .37g horizontal and .25g vertical to insure the integrity of the lockout relay.

All monitored circuits performed as required during the sine dwell test.

c) Circuit monitoring

Following the reinforcement and bolting of the rear door no further relay chatter occurred.

d) Insulation resistance

The insulation resistance measurements were taken from a terminal point to ground (readings in megohms):

Circuit Tested	Before Test	After Test
100	20	17.5
102	21	17.0

CONCLUSION

To insure the reliability of the structure addition formed gussets and/or channels were added to each compartment with emphasis on directing the stress toward the base of the cabinit. This elevated the natural frequency a greater amount over the floor spectra.

APPENDIX 3.J

ALTERNATIVE POSTULATED RUPTURE LOCATIONS IN FLUID SYSTEM PIPING INSIDE AND OUTSIDE CONTAINMENT FOR ASME CLASS 2, CLASS 3 AND NON ASME CLASS PIPING SYSTEMS

BRANCH TECHNICAL POSITION MEB 3-1
AS APPLICABLE TO ST. LUCIE UNIT 1

3.6-1
APPENDIX 3.J

POSTULATED RUPTURE LOCATIONS IN FLUID SYSTEM PIPING INSIDE AND OUTSIDE CONTAINMENT

(BRANCH TECHNICAL POSITION MEB 3-1)

A. BACKGROUND

This position on pipe rupture postulation is intended to comply with the requirements of General Design Criterion 4, of Appendix A to 10 CFR Part 50 for the design of nuclear power plant structures and components. It is recognized that pipe rupture is a rare event which may only occur under unanticipated conditions, such as those which might be caused by possible design, construction, or operation errors; unanticipated loads or unanticipated corrosive environments. Our observation of actual piping failures has indicated that they generally occur at high stress and fatigue locations, such as at the terminal ends of a piping system at its connection to the nozzles of a component. The rules of this position are intended to utilize the available piping design information by postulating pipe ruptures at locations having relatively higher potential for failure, such that an adequate and practical level of protection may be achieved.

B. DESIGN CRITERIA FOR THE DETERMINATION OF POSTULATED PIPING BREAKS

1. High-Energy Fluid Systems Piping

a. Fluid Systems Separated From Essential Systems and Components

For the purpose of satisfying the separation provisions of plant arrangement as specified in B.1.a of Branch Technical Position (BTP) ASB 3-1, a review of the piping layout and plant arrangement drawings should clearly show the effects of postulated piping breaks at any location are isolated or physically remote from essential systems and components[1]. At the designer's option, break locations as determined from B.1.c. of this criteria may be assumed for this purpose.

b. Fluid System Piping in Containment Penetration Areas

Breaks and cracks need not be postulated in those portions of piping from containment wall to and including the inboard or outboard isolation valves provided they meet the requirement of the ASME Code, Section III, Subarticle NE-1120 and the following additional design requirements:

[1] Systems and components required to shut down the reactor and mitigate the consequences of a postulated pipe rupture without offsite power.

B.1.b. (1) The following design stress and fatigue limits should not be exceeded:

For ASME Code, Section III, Class 1 Piping

(a), (b), (c) THESE SECTIONS ARE INTENTIONALLY OMITTED

For ASME Code, Section III, Class 2 Piping

(d) The maximum stress as calculated by the sum of Eqs. (9) and (10) in Paragraph NC-3652, ASME Code, Section III, considering those loads and conditions thereof for which level A and level B stress limits have been specified In the system's Design Specification (i.e., sustained loads, occasional loads, and thermal expansion) including an OBE event should not exceed $0.8(1.8\ S_h + S_A)$. The S_h and S_A are allowable stresses at maximum (hot) temperature and allowable stress range for thermal expansion, respectively, as defined in Article NC-3600 of the ASME Code, Section III.

(e) The maximum stress, as calculated by Eq. (9) in NC-3653 under the loadings resulting from a postulated piping failure of fluid system piping beyond these portions of piping should not exceed the lesser of $2.25\ S_h$ and $1.8\ S_Y$.

Primary loads include those which are deflection limited by whip restraints. Higher Stresses may be allowed following a failure when the following conditions exist:

i) The piping is between the outboard isolation valve and the first restraint
ii) A plastic hinge is not formed
iii) operability of the valves with the higher stress is assured in accordance with the requirements specified in SRP Section 3.9.3
iv) piping is constructed in accordance with ANSI B31.1
v) The piping shall either be of seamless construction with full radiography of all circumferential welds, or all longitudinal and circumferential welds shall be fully radiographed.

(2) Welded attachments, for pipe supports or other purposes, to these portions of piping should be avoided except where detailed stress analyses, or tests, are performed to demonstrate compliance with the limits of B.1.b.(1).

(3) The number of circumferential and longitudinal piping welds and branch connections should be minimized. Where guard pipes are used, the enclosed portion of <u>fluid system</u> piping should be seamless construction and without circumferential welds unless specific access provisions are made to permit inservice volumetric examination of the longitudinal and circumferential welds.

(4) The length of these portions of piping should be reduced to the minimum length practical.

B.1.b (5) The design of pipe anchors or restraints (e.g., connections to containment penetrations and pipe whip restraints) should not require welding directly to the outer surface of the piping (e.g., flued integrally forged pipe fittings may be used) except where such welds are 100 percent volumetrically examinable in service and a detailed stress analysis is performed to demonstrate compliance with the limits of B.1.b.(1).

(6) Guard pipes provided for those portions of piping in the containment penetration areas should be constructed in accordance with the rules of Class MC, Subsection NE of the ASME Code, Section III, where the guard pipe is part of the containment boundary. In addition, the entire guard pipe assembly should be designed to meet the following requirements and tests:

(a) The design pressure and temperature should not be less than the maximum operating pressure and temperature of the enclosed pipe under <u>normal plant conditions</u>.

(b) The Level C stress limits in NE-3220, ASME Code, Section III, should not be exceeded under the loadings associated with containment design pressure and temperature in combination with the safe shutdown earthquake.

(c) Guard pipe assemblies should be subjected to a single pressure test at a pressure not less than its design pressure.

(d) Guard pipe assemblies should not prevent the access required to conduct the inservice examination specified in B.1.b.(7). Inspection ports, if used, should not be located in that portion of the guard pipe through the annulus of dual barrier containment structures.

(7) A 100% volumetric inservice examination of all pipe welds should be conducted during each inspection interval as defined in IWA-2400, ASME Code, Section XI.

B.1.c. <u>Postulation of Pipe Breaks In Areas Other Than Containment Penetration</u>

(1) THIS SECTION INTENTIONALLY OMITED

(2) With the exceptions of those portions of piping identified in B.1.b., breaks in Class 2 and 3 piping (ASME Code, Section III) should be postulated at the following locations in those portions of each piping and branch run:

(a) At <u>terminal ends</u>.

(b) At intermediate locations selected by one of the following criteria:

(i) At each pipe fitting (e.g., elbow, tee, cross, flange, and nonstandard fitting), welded attachment, and valve. Where the piping contains no fittings, welded attachments, or valves, at one location at each extreme of the piping run adjacent to the protective structure.

B.1.c (2) (b) (ii) At each location where stresses calculated[2] by the sum of Eqs. (9) and (10) in NC/ND-3653, ASME Code, Section III, exceed 0.8 times the sum of the stress limits given in NC/ND-3653.

As a result of piping reanalysis due to differences between the design configuration and the as-built configuration, the highest stress locations may be shifted; however, the initially determined intermediate break locations may be used unless a redesign of the piping resulting in a change in pipe parameters (diameter, wall thickness, routing) is required, or the dynamic effects from the new (as-built) intermediate break locations are not mitigated by the original pipe whip restraints and jet shields.

(3) Breaks in seismically analyzed non-ASME Class piping are postulated according to the same requirements for ASME Class 2 and 3 piping above[3].

(4) Applicable to (1), (2) and (3) above:

If a structure separates a high energy line from an essential component, that separating structure should be designed to withstand the consequences of the pipe break in the high energy line which produces the greatest effect at the structure irrespective of the fact that the above criteria might not require such a break location to be postulated.

(5) Safety-related equipment must be environmentally qualified in accordance with 10 CFR 50.49. Required pipe ruptures and leakage cracks (whichever controls) must be included in the design bases for environmental qualification of electrical and mechanical equipment both inside and outside the containment.

B.1.d. The designer should identify each piping run he has considered to postulate the break locations required by B.1.c. above. In complex systems such as those containing arrangements of headers and parallel piping running between headers, the designer should identify and include all such piping within a designated run in order to postulate the number of breaks required by these criteria.

B.1.e. With the exception of those portions of piping identified in B.1.b, leakage cracks should be postulated as follows:

(1) THIS SECTION INTENTIONALLY OMITTED

(2) For ASME Code, Section III Class 2 and 3 or nonsafety class (not ASME Class 1, 2 or 3) piping, at axial locations where the calculated stress[2] by the sum of Eqs. (9) and (10) in NC/ND-3653 exceeds 0.4 times the sum of the stress limits given tn NC/ND-3653.

---

[2] For those loads and conditions in which Level A and Level B stress limits have been specified in the Design Specification (including the operating basis earthquake).

[3] Note that in addition, breaks in non-seismic, that is, non-Category I piping, are to be taken into account as described in Section II.2.k. "Interaction of Other Piping with Category I Piping" of SRP 3.9.2.

B.1.e. (3) Nonsafety class piping which has not been evaluated to obtain stress information should have leakage cracks postulated to axial locations that produce the most severe environmental effects.

B.2. Moderate-Energy Fluid System Piping

The design bases for St. Lucie Unit 1 required only High Energy piping systems to be analyzed and protected for postulated pipe breaks. In some cases, portions of systems with pressures as low as 125 psig were analyzed and protected for pipe whip consistent with the same methodology as high energy piping systems.

Those portions of systems as low as 125 psig shall be analyzed in accordance with the original plant criteria or section B.1 above. THEREFORE, THE MODERATE ENERGY CRITERIA PER MEB 3-1 IS INTENTIONALLY OMITTED.

B.3. Type of Breaks and Leakage Cracks in Fluid System Piping

B.3.a. Circumferential Pipe Breaks

The following circumferenttal breaks should be postulated individually in high-energy fluid system piping at the locations spectfied in B.1 of this criteria:

(1) Circumferential breaks should be postulated in fluid system piping and branch runs exceeding a nominal pipe size of 1 inch, except where the maximum stress range[2] exceeds the limits specified in B.1.c.(1) and B.1.c.(2) but the circumferential stress range is at least 1.5 times the axial stress range. Instrument lines, one inch and less nominal pipe or tubing size should meet the provisions of Regulatory Guide 1.11.

(2) Where break locations are selected without the benefit of stress calculations, breaks should be postulated at the piping welds to each fitting, valve, or welded attachment.

(3) Circumferential breaks should be assumed to result in pipe severance and separation amounting to at least a one-diameter lateral displacement of the ruptured piping sections unless physically limited by piping restraints, structural members, or piping stiffness as may be demonstrated by inelastic limit analysis (e.g., a plastic hinge in the piping is not developed under loading).

(4) The dynamic force of the jet discharge at the break location should be based on the effective cross-sectional flow area of the pipe and on a calculated fluid pressure as modified by an analytically or experimentally determined thrust coefficient. Limited pipe displacement at the break location, line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account, as applicable, in the reduction of jet discharge.

(5) Pipe whipping should be assumed to occur in the plane defined by the piping geometry and configuration, and to initiate pipe movement in the direction of the jet reaction.

B.3.b. Longitudinal Pipe Breaks

The following longitudinal breaks should be postulated in high-energy fluid system piping at the locations of the circumferential breaks specified in B.3.a:

(1) Longitudinal breaks in fluid system piping and branch runs should be postulated in nominal pipe sizes 4-inch and larger, except where the maximum stress range exceeds the limits specified in B.1.c.(1) and B.1.c.(2) but the axial stress range is at least 1.5 times the circumferential stress range.

(2) Longitudinal breaks need not be postulated at terminal ends.

(3) Longitudinal breaks should be assumed to result in an axial split without pipe severance. Splits should be oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that the jet reactions cause out-of-plane bending of the piping configuration. Alternatively, a single split may be assumed at the section of highest tensile stress as determined by detailed stress analysis (e.g., finite element analysis).

(4) The dynamic force of the fluid jet discharge should be based on a circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified byan analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account as applicable, in the reduction of jet discharge.

(5) Piping movement should be assumed to occur in the direction of the jet reaction unless limited by structural members, piping restraints, or piping stiffness as demonstrated by inelastic limit analysis.

B.3.c. Leakage Crack

Leakage cracks should be postulated at those axial locations specified in B.1.e for high-energy fluid system piping and in those piping systems not exempted in B.2.c.(1) for moderate-energy fluid system piping.

(1) Leakage cracks need not be postulated in 1 inch and smaller piping.

(2) For high-energy fluid system piping, the leakage cracks should be postulated to be in those circumferential locations that result in the most severe environmental consequences. For moderate-energy fluid system piping, see B.2.c.(2).

(3) Fluid flow from a leakage crack should be based on a circular opening of area equal to that of a rectangle one-half pipe diameter in length and one-half pipe wall thickness in width.

(4) The flow from the leakage crack should be assumed to result in an environment that wets all unprotected components within the compartment, with consequent flooding in the compartment and communicating compartments. Flooding effects should be determined on the basis of a conservatively estimated time period required to effect corrective actions.

C. REFERENCES

1. 10 CFR Part 50, Appendix A, General Design Criterion 4, "Environmental and Missile Design Basis."

2. "Boiler and Pressure Vessel Code," Sections III and XI, American Society of Mechanical Engineers, 1986 Edition.

3. Regulatory Guide 1.11, "Instrument Lines Penetrating Primary Reactor Containment."