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Attn: Mr. W. T. Russell

July 19, 1994

Dear Mr. Russell:

Attached are ten copies of the changed pages for the 1994 update to the Cook Nuclear Plant Final Safety Analysis Report. These pages are being transmitted to you according to the provisions of 10 CFR 50.71(e). Instructions for incorporating the update are included with each copy.

Changed pages have been dated "July, 1994" in the lower right corner in order to identify changed pages in addition to vertically barring the specific change. Vertical change bars next to the July 1994 date in the lower right corner indicate that the information has only shifted pages.

We hereby certify that the information contained in this update to the FSAR, to our knowledge, accurately presents changes made between January 22, 1993, and January 22, 1994.

Sincerely,

E E Fitzpatrick

E. E. Fitzpatrick
Vice President

af

Attachment

A053 1/10

cc: w/o attachment
A. A. Blind - Bridgman
G. Charnoff
J. B. Martin
J. R. Padgett
NRC Resident Inspector - Bridgman
NFEM Section Chief

VOLUME I

Chapter 1	Introduction and Summary	Page	Date
		1.0-1	1988
		1.0-2	1989
		1.0-3	1989
		1.1-1	1982
		1.1-2	1982
		1.1-3	1982
		1.1-4	1982
		Fig. 1-1	ORIG
		1.2-1	1993
		1.2-2	1993
		1.2-3	1993
		1.2-4	1991
		1.2-5	1982
		Table 1.2-1 (6 pages)	1989
		1.3-1	1982
		1.3-2	1982
		1.3-3	1982
		1.3-4	1984
		1.3-5	1982
		1.3-6	1982
		1.3-7	1982
		1.3-8	1984
		1.3-9	1982
		Fig. 1.3-1	1993
		Fig. 1.3-2	1990
		Fig. 1.3-3	1990
		Fig. 1.3-4	1990
		Fig. 1.3-5	1990
		Fig. 1.3-6	1990
		Fig. 1.3-7	1990
		Fig. 1.3-8	1993
		Fig. 1.3-9	1993
		Fig. 1.3-10	1993
		Fig. 1.3-11	1982
		1.4-1	1991
		1.4-2	1991
		1.4-3	1987
		1.4-4	1987
		1.4-5	1991
		1.4-6	1982
		1.4-7	1991
		1.4-8	1991
		1.4-9	1982
		1.4-10	1987
		1.4-11	1991

Superseded pages Rev 1994 UFSAR 940076

50-315/316

9407260242

7/19/94

VOLUME I

Chapter 1	Introduction and Summary	Page	Date
		1.4-12	1991
		1.4-13	1991
		1.4-14	1991
		1.4-15	1991
		1.4-16	1993
		1.4-17	1987
		1.4-18	1987
		1.4-19	1992
		1.4-20	1992
		1.4-21	1991
		1.4-22	1991
		Table 1.4-1 (pg 1)	1991
		(pg 2)	1991
		(pg 3)	1991
		(pg 4)	1991
		1.5-1	1982
		1.6-0	1984
		1.6-1	1983
		1.6-2	1983
		1.6-3	1982
		1.6-4	1982
		1.6-5	1983
		1.6-6	1982
		1.6-7	1983
		1.6-8	1982
		1.6-9	1982
		1.6-10	1983
		1.6-11	1982
		1.6-12	1983
		1.6-13	1982
		1.6-14	1985
		1.6-15	1985
		1.6-16	1985
		1.6-17	1985
		1.6-18	1985
		1.6-19	1992
		1.6-20	1985
		1.6-21	1985
		1.6-22	1985
		1.6-23	1985
		1.6-24	1985
		1.6-25	1992
		Table 1.6-1 (pg 1)	1989
		(pg 2)	1989
		(pg 3)	1989
		(pg 4)	1989
		(pg 5)	1989
		(pg 6)	1989
		(pg 7)	1989
		(pg 8)	1989
		(pg 9)	1989
		(pg 10)	1989

VOLUME I

VOLUME I

Chapter 2 Site and Environment	Page	Date
	2.1-1	1991
	2.1-2	1993
	2.1-3	1993
	2.1-4	1982
	2.1-5	1993
	2.1-6	1993
	2.1-7	1993
	2.1-8	1993
	2.1-9	1993
	2.1-10	1982
	2.1-11	1993
	2.1-12	1993
	Table 2.1-1	1989
	Table 2.1-2	1989
	Table 2.1-3	1989
	Table 2.1-4	1993
	Table 2.1-5	1993
	Table 2.1-6	1993
	Table 2.1-7	1993
	Table 2.1-8	1993
	Table 2.1-9	1993
	Table 2.1-10	1993
	Table 2.1-11	1993
	Table 2.1-12	1993
	Fig. 2.1-1	1982
	Fig. 2.1-2	1982
	Fig. 2.1-3	1993
	Fig. 2.1-4	1982
	Fig. 2.1-4a	1982
	Fig. 2.1-4b	1993
	Fig. 2.1-5	1993
	Fig. 2.1-6	1982
	Fig. 2.1-7	1993
	Fig. 2.1-8	1993
	Fig. 2.1-9	1993
	Fig. 2.1-10	1993
	2.2-1	1993
	2.2-2	1993
	2.2-3	1993
	2.2-4	1993
	2.2-5	1988
	2.2-6	1993
	2.2-7	1993

VOLUME I

Chapter 2 Site and Environment	Page	Date
	2.2-8	1993
	2.2-9	1993
	2.2-10	1993
	Table 2.2-1	1993
	Table 2.2-2	1993
	Table 2.2-3	1993
	Table 2.2-4	1993
	Table 2.2-5	1993
	Table 2.2-6	1993
	Table 2.2-7	1993
	Table 2.2-8	1993
	Table 2.2-9	1993
	Table 2.2-10	Deleted
	Table 2.2-11	Deleted
	Table 2.2-12	Deleted
	Table 2.2-13	Deleted
	Table 2.2-14	Deleted
	Table 2.2-15	Deleted
	Table 2.2-16	Deleted
	Table 2.2-17	Deleted
	Table 2.2-18	Deleted
	Table 2.2-19	Deleted
	Table 2.2-20	Deleted
	Table 2.2-21	Deleted
	Table 2.2-22	Deleted
	Table 2.2-23	Deleted
	Table 2.2-24	Deleted
	Table 2.2-25	Deleted
	Table 2.2-26	Deleted
	Table 2.2-27	Deleted
	Table 2.2-28	Deleted
	Table 2.2-29	Deleted
	Table 2.2-30	Deleted
	Table 2.2-31	Deleted
	Table 2.2-32	Deleted
	Table 2.2-33	Deleted
	Fig. 2.2-1	1982
	Fig. 2.2-2	1993
	Fig. 2.2-3	1993
	Fig. 2.2-4	1993
	Fig. 2.2-5	1993
	Fig. 2.2-6	1993

VOLUME I

VOLUME I

VOLUME I

VOLUME II

VOLUME II

Chapter 3	Reactor (Unit 1)	Page	Date
		3.2-32	1982
		3.2-33	1982
		3.2-34	1983
		3.2-35	1982
		3.2-36	1982
		3.2-37	1982
		3.2-38	1982
		3.2-39	1982
		3.2-40	1982
		3.2-41	1982
		3.2-42	1982
		3.2-43	1982
		3.2-44	1983
		3.2-45	1982
		3.2-46	1987
		3.2-47	1987
		3.2-48	1987
		3.2-49	1987
		3.2-50	1982
		Table 3.2.1-1 (3pp)	1989
		Fig. 3.2.1-1	1982
		Fig. 3.2.1-2	1982
		Fig. 3.2.1-3	1982
		Fig. 3.2.1-4	1982
		Fig. 3.2.1-5	1982
		Fig. 3.2.1-6	1982
		Fig. 3.2.1-7	1982
		Fig. 3.2.1-8	1982
		Fig. 3.2.1-9	1982
		Fig. 3.2.1-10	1982
		Fig. 3.2.1-11	1982
		Fig. 3.2.1-12	1982
		Fig. 3.2.1-13	1982
		Fig. 3.2.1-14	1982
		3.3-1	1990
		3.3-2	1992
		3.3-3	1992
		3.3-4	1982
		3.3-5	1992
		3.3-6	1992
		3.3-7	1983
		3.3-8	1983
		3.3-9	1984
		3.3-10	1990
		3.3-11	1987
		3.3-12	1982

Chapter 3	Reactor (Unit 1)	Page		Date
		3.3-13		1982
		3.3-14		1992
		3.3-15		1992
		3.3-16		1992
		3.3-17		1992
		3.3-18		1992
		3.3-19		1992
		3.3-20		1992
		3.3-21		1992
		3.3-22		1992
		3.3-23		1992
		3.3-24		1992
		3.3-25		1993
		3.3-26		1989
		3.3-27		1990
		Table 3.3.1-1	(pg 1)	1992
			(pg 2)	1990
			(pg 3)	1993
		Table 3.3.1-2		1990
		Table 3.3.1-3		1993
		Table 3.3.1-3a		1990
		Fig. 3.3.1-1		1982
		Fig. 3.3.1-2		1984
		Fig. 3.3.1-3		1984
		Fig. 3.3.1-4		1984
		Fig. 3.3.1-5		DELETED
		Fig. 3.3.1-6		DELETED
		Fig. 3.3.1-7		DELETED
		Fig. 3.3.1-8		DELETED
		Fig. 3.3.1-9		DELETED
		Fig. 3.3.1-10		DELETED
		Fig. 3.3.1-11		1984
		Fig. 3.3.1-12		1984
		Fig. 3.3.1-13		1984
		Fig. 3.3.1-14		1984
		Fig. 3.3.1-15		1984
		Fig. 3.3.1-16		1984
		Fig. 3.3.1-17		1992
		3.4-1		1990
		3.4-2		1993
		3.4.3		1993
		3.4.4		1982
		3.4.5		1982
		3.4-6		1982
		3.4-7		1993

Chapter 3	Reactor (Unit 1)	Page	Date
		3.4-8	1993
		3.4-9	1993
		3.4-10	1982
		3.4-11	1993
		3.4-12	1987
		3.4-13	1993
		3.4-14	1993
		3.4-15	1983
		3.4-16	1982
		3.4-17	1982
		3.4-18	1982
		3.4-19	1982
		3.4-20	1982
		3.4-21	1982
		Table 3.4.1-1 (2pp)	1989
		Table 3.4.1-2	1989
		Table 3.4.1-3	1989
		Fig. 3.4.1-1	1982
		Fig. 3.4.1-2	1982
		Fig. 3.4.1-3	1982
		Fig. 3.4.1-4	1982
		Fig. 3.4.1-4a	1982
		Fig. 3.4.1-5	1982
		Fig. 3.4.1-6	1982
		Fig. 3.4.1-7	1982
		Fig. 3.4.1-8	1982
		Fig. 3.4.1-9	1982
		3.5-1	1993
		3.5.1-1	1993
		3.5.1-2	1993
		3.5.1-3	1993
		3.5.1-4	1993
		3.5.1-5	1993
		3.5.1-6	1993
		3.5.1-7	1993
		3.5.1-8	1993
		3.5.1-9	1993
		3.5.1-10	1993
		3.5.1-11	1993
		3.5.1-12	1993
		3.5.1-13	1993
		3.5.1-14	1993
		3.5.1-15	1993
		3.5.1-16	1993
		3.5.1-17	1990
		3.5.1-18	1993

<u>VOLUME II</u>

Chapter 3	Reactor (Unit 1)	Page	Date
		Table 3.5.1-1	1992
		Table 3.5.1-2	Deleted
		Table 3.5.1-3	Deleted
		Fig. 3.5.1-1	1990
		Fig. 3.5.1-2	1990
		Fig. 3.5.1-3	1990
		Fig. 3.5.1-4	1990
		Fig. 3.5.1-5	1990
		Fig. 3.5.1-5a	1992
		Fig. 3.5.1-6	1992
		Fig. 3.5.1-7	1992
		Fig. 3.5.1-8	Deleted
		FIg. 3.5.1-9	Deleted
		3.5.2-1	1992
		3.5.2-2	1992
		3.5.2-3	1993
		3.5.2-4	1993
		3.5.2-5	1992
		3.5.2-6	1993
		Table 3.5.2-1	1993
		Table 3.5.2-2	1991
		Table 3.5.2-3	1993
		Fig. 3.5.2-1	1993
		Fig. 3.5.2-2	1992
		3.5.3-1	1993
		3.5.3-2	1992
		3.5.3-3	1990
		3.5.3-4	1990
		3.5.3-5	1992
		3.5.3-6	1992
		3.5.3-7	1993
		3.5.3-8	1990
		3.5.3-9	1992
		3.5.3-10	1993
		3.5.3-11	1990
		3.5.3-12	1990
		3.5.3-13	1992
		3.5.3-14	1991
		Table 3.5.3-1 (pg 1)	1992
		Table 3.5.3-1 (pg 2)	1992
		Table 3.5.3-1 (pg 3)	1990
		Table 3.5.3-1 (pg 4)	1992
		Fig. 3.5.3-1	1990
		Fig. 3.5.3-2	1990
		Fig. 3.5.3-3	1990

VOLUME II

Chapter 3	Reactor (Unit 2)	Page	Date
		3.2-36	1991
		3.2-37	1991
		3.2-38	1991
		3.2-39	1991
		3.2-40	1991
		3.2-41	1991
		3.2-42	1991
		3.2-43	1991
		3.2-44	1991
		3.2-45	1991
		3.2-46	1991
		3.2-47	1991
		3.2-48	1991
		3.2-49	1991
		3.2-50	1991
		3.2-51	1991
		3.2-52	1991
		3.2-53	1991
		3.2-54	1991
		3.2-55	1991
		3.2-56	1991
		3.2-57	1991
		3.2-58	1991
		3.2-59	1991
		3.2-60	1982
		3.2-61	1982
		3.2-62	1982
		3.2-63	1982
		3.2-64	1982
		3.2-65	1982
		3.2-66	1982
		3.2-67	1991
		3.2-68	1982
		3.2-69	1982
		3.2-70	1982
		3.2-71	1982
		3.2-72	1982
		3.2-73	1991
		3.2-74	1982
		3.2-75	1982
		3.2-76	1991
		3.2-77	1982
		3.2-78	1982
		3.2-79	1982
		3.2-80	1982
		3.2-81	1982
		3.2-82	1991
		3.2-83	1991
		3.2-84	1991
		3.2-85	1982

VOLUME II

Chapter 3	Reactor (Unit 2)	Page	Date
		3.2-86	1982
		3.2-87	1982
		3.2-88	1991
		3.2-89	1991
		3.2-90	1991
		Table 3.2-1	1991
		Fig. 3.2-1	1991
		Fig. 3.2-2	1991
		Fig. 3.2-3	1991
		Fig. 3.2-4	1991
		Fig. 3.2-5	1991
		Fig. 3.2-5a	1991
		Fig. 3.2-6	1991
		Fig. 3.2-7	1991
		Fig. 3.2-8	1982
		Fig. 3.2-9	1982
		Fig. 3.2-10	1982
		Fig. 3.2-11	1982
		Fig. 3.2-12	1982
		Fig. 3.2-13	1982
		Fig. 3.2-14	DELETED
		Fig. 3.2-15	1982
		Fig. 3.2-16	1982
		Fig. 3.2-17	1982
		Fig. 3.2-18	1990
		Fig. 3.2-19	1982
		Fig. 3.2-20	1982
		Fig. 3.2-21	1982
		Fig. 3.2-22	1982
		Fig. 3.2-23	1982
		Fig. 3.2-24	1982

<u>VOLUME III</u>

Chapter 3	Reactor (Unit 2)	Page	Date
		3.3-1	1991
		3.3-2	1991
		3.3-3	1991
		3.3-4	1992
		3.3-5	1991
		3.3-6	1991
		3.3-7	1991
		3.3-8	1991
		3.3-9	1993
		3.3-10	1991
		3.3-11	1991
		3.3-12	1991
		3.3-13	1991
		3.3-14	1991
		3.3-15	1991
		3.3-16	1991
		3.3-17	1991
		3.3-18	1991
		3.3-19	1991
		3.3-20	1991
		3.3-21	1991
		3.3-22	1991
		3.3-23	1992
		3.3-24	1991
		3.3-25	1991
		3.3-26	1991
		3.3-27	1991
		3.3-28	1991
		3.3-29	1991
		3.3-30	1991
		3.3-31	1991
		3.3-32	1991
		3.3-33	1991
		3.3-34	1991
		3.3-35	1993
		3.3-36	1991
		3.3-37	1991
		3.3-38	1991
		3.3-39	1991
		3.3-40	1991
		3.3-41	1991
		3.3-42	1991
		3.3-43	1991
		3.3-44	1991
		3.3-45	1991
		3.3-46	1991
		3.3-47	1991

VOLUME III

VOLUME III

Chapter 3	Reactor (Unit 2)	Page	Date
		Fig. 3.3-11	1991
		Fig. 3.3-12	1991
		Fig. 3.3-13	1991
		Fig. 3.3-14	1991
		Fig. 3.3-15	1991
		Fig. 3.3-16	1991
		Fig. 3.3-17	1991
		Fig. 3.3-18	1991
		Fig. 3.3-19	1991
		Fig. 3.3-20	1991
		Fig. 3.3-21	1991
		Fig. 3.3-22	1991
		Fig. 3.3-23	1991
		Fig. 3.3-24	1991
		Fig. 3.3-25	1991
		Fig. 3.3-26	1991
		Fig. 3.3-27	1991
		Fig. 3.3-28	1991
		Fig. 3.3-29	1991
		Fig. 3.3-30	1991
		Fig. 3.3-31	1991
		Fig. 3.3-32	1991
		Fig. 3.3-33	1991
		Fig. 3.3-34	1991
		Fig. 3.3-35	1991
		Fig. 3.3-36	1991
		Fig. 3.3-37	1991
		Fig. 3.3-38	1991
		3.4-1	1982
		3.4-2	1991
		3.4-3	1991
		3.4-4	1991
		3.4-5	1993
		3.4-6	1991
		3.4-7	1991
		3.4-8	1991
		3.4-9	1991
		3.4-10	1992
		3.4-11	1991

VOLUME III

Chapter 3	Reactor (Unit 2)	Page	Date
		3.4-12	1991
		3.4-13	1991
		3.4-14	1991
		3.4-15	1993
		3.4-16	1991
		3.4-17	1991
		3.4-18	1991
		3.4-19	1991
		3.4-20	1991
		3.4-21	1991
		3.4-22	1991
		3.4-23	1993
		3.4-24	1991
		3.4-25	1993
		3.4-26	1991
		3.4-27	1991
		3.4-28	1991
		3.4-29	1991
		3.4-30	1993
		3.4-31	1991
		3.4-32	1993
		3.4-33	1991
		3.4-34	1991
		3.4-35	1991
		3.4-36	1991
		3.4-37	1991
		3.4-38	1991
		3.4-39	1991
		3.4-40	1993
		3.4-41	1991
		3.4-42	1991
		3.4-43	1992
		3.4-44	1991
		3.4-45	1991
		3.4-46	1993
		3.4-47	1991
		3.4-48	1991
		3.4-49	1991
		3.4-50	1991
		3.4-51	1992
		3.4-52	1991
		3.4-53	1991
		3.4-54	1991
		3.4-55	1991
		3.4-56	1991
		3.4-57	1991
		3.4-58	1991
		3.4-59	1991
		3.4-60	1991

VOLUME III

Chapter 3	Reactor (Unit 2)	Page	Date
		3.4-61	1991
		3.4-62	1991
		3.4-63	1991
		3.4-64	1991
		3.4-65	1991
		Table 3.4-1 (pg 1)	1992
		(pg 2)	1992
		(pg 3)	1991
		Table 3.4-2	1991
		Table 3.4-3 (2pp)	1991
		Table 3.4-4	1991
		Fig. 3.4-1	1991
		Fig. 3.4-2	1991
		Fig. 3.4-3	1991
		Fig. 3.4-4	1991
		Fig. 3.4-5	1991
		Fig. 3.4-6	1991
		Fig. 3.4-7	1991
		Fig. 3.4-8	1991
		Fig. 3.4-9	1991
		Fig. 3.4-10	1991
		Fig. 3.4-11	1991
		Fig. 3.4-12	1991
		Fig. 3.4-13	1991
		Fig. 3.4-14	1991
		3.5-1	1991
		3.5-2	1987
		3.5-3	1991
		3.5-4	1991
		3.5-5	1991
		3.5-6	1991
		3.5-7	1991
		3.5-8	1991
		3.5-9	1991
		3.5-10	1993
		3.5-11	1991

VOLUME III

Chapter 3	Reactor (Unit 2)	Page	Date
		3.5-12	1991
		3.5-13	1991
		3.5-14	1991
		3.5-15	1991
		3.5-16	1991
		3.5-17	1991
		3.5-18	1991
		3.5-19	1987
		3.5-20	1990
		Table 3.5.1-1 (2pp)	1991
		Table 3.5.1-2 (2pp)	1991
		Table 3.5.1-3	1991
		Fig. 3.5.1-1	1983
		Fig. 3.5.1-2	ORIG
		Fig. 3.5.1-3	1991
		Fig. 3.5.1-4	1983
		3.5-26	1991
		3.5-27	1991
		3.5-28	1991
		Table 3.5.2-1	1991

VOLUME III

Chapter 4	Reactor Coolant System	Page	Date
		4.1-1	1991
		4.1-2	1991
		4.1-3	1982
		4.1-4	1982
		4.1-5	1982
		4.1-6	1982
		4.1-7	1982
		4.1-8	1982
		4.1-9	1982
		4.1-10	1982
		4.1-11	1982
		4.1-12	1990
		4.1-13	1990
		4.1-14	1990
		4.1-15	1990
		4.1-16	1991
		4.1-17	1982
		4.1-18	1982
		4.1-19	1989
		4.1-20	1982
		4.1-21	1982
		4.1.22	1982
		4.1-23	1982
		4.1-24	1982
		Table 4.1-1	1990
		Table 4.1-2	1993
		Table 4.1-3	1990
		Table 4.1-4	1991
		Table 4.1-5 (pg 1)	1991
		(pg 2)	1989
		Table 4.1-6	1989
		Table 4.1-7	1989
		Table 4.1-8	1989
		Table 4.1-9	1989
		Table 4.1-10 (pg 1)	1990
		(pg 2)	1989
		Table 4.1-11 (3pp)	1989
		Table 4.1-12 (2pp)	1991
		4.2-1	1982
		4.2-2	1982
		4.2-3	1982
		4.2-4	1989
		4.2-5	1982
		4.2-6	1982
		4.2-7	1982
		4.2-8	1982
		4.2-9	1982
		4.2-10	1982
		4.2-11	1989
		4.2-12	1989

VOLUME III

Chapter 4	Reactor Coolant System	Page	Date
		4.2-13	1989
		4.2-14	1982
		4.2-15	1982
		4.2-16	1982
		4.2-17	1990
		4.2-18	1983
		4.2-19	1991
		4.2-20	1991
		4.2-21	1991
		4.2-22	1993
		4.2-23	1982
		4.2-24	1982
		4.2-25	1982
		4.2-26	1986
		4.2-27	1982
		4.2-28	1986
		4.2-29	1982
		4.2-30	1982
		4.2-31	1982
		4.2-32	1982
		4.2-33	1982
		4.2-34	1987
		4.2-35	1987
		4.2-36	1987
		Table 4.2-1 (3pp)	1989
		Table 4.2-2	1992
		Table 4.2-3	1993
		Fig. 4.2-1	1984
		Fig. 4.2-1A	1988
		Fig. 4.2-2	1982
		Fig. 4.2-2A	1982
		Fig. 4.2-3	1982
		Fig. 4.2-4	1982
		Fig. 4.2-4A	1989
		Fig. 4.2-5	1982
		Fig. 4.2-6	1982
		Fig. 4.2-7	1982
		Fig. 4.2-8	1982
		Fig. 4.2-9	1982
		Fig. 4.2-9 Ref. (4pp)	1982
		4.3-1	1982
		4.3-2	1982
		4.3-3	1982
		4.3-4	1982
		4.3-5	1982
		4.3-6	1982
		4.3-7	1990
		4.3-8	1982
		4.3-9	1982

Chapter 4	Reactor Coolant System	Page	Date
		4.3-10	1982
		4.3-11	1989
		4.3-12	1989
		4.3-13	1989
		4.3-14	1989
		4.3-15	1989
		4.3-16	1989
		4.3-17	1989
		4.3-18	1989
		4.3-19	1989
		4.3-20	1989
		4.3-21	1989
		4.3-22	1989
		4.3-23	1989
		4.3-24	1989
		4.3-25	1989
		4.3-26	1989
		Table 4.3-1	1990
		Table 4.3-2	1990
		Table 4.3-3	1989
		Table 4.3-4	1989
		Table 4.3-5 (2pp)	1990
		Table 4.3-6	1990
		Table 4.3-7	1990
		Table 4.3-8	1989
		Fig. 4.3-1	1982
		Fig. 4.3-2	1982
		Fig. 4.3-3	1982
		Fig. 4.3-4	1982
		Fig. 4.3-5	1982
		Fig. 4.3-6	1990
		Fig. 4.3-7	1990
		4.4-1	1986
		4.4-2	1988
		4.4-3	1986
		4.5-1	1982
		4.5-2	1988
		4.5-3	1982
		4.5-4	1982
		4.5-5	1993
		4.5-6	1993
		4.5-7	1982
		4.5-8	1982
		4.5-9	1982
		4.5-10	1982
		4.5-11	1982
		4.5-12	1982

Chapter 5	Containment System	Page	Date
		5.0-1	1989
		5.1-1	1987
		5.1-2	1982
		5.1-3	1982
		5.1-4	1982
		Table 5.1-1 (2pp)	1989
		5.2-1	1989
		5.2-2	1982
		5.2-3	1982
		5.2-4	1982
		5.2-5	1982
		5.2-6	1986
		5.2-7	1982
		5.2-8	1990
		5.2-9	1986
		5.2-10	1982
		5.2-11	1982
		5.2-12	1982
		5.2-13	1982
		5.2-14	1982
		5.2-15	1982
		5.2-16	1982
		5.2-17	1982
		5.2-18	1982
		5.2-19	1982
		5.2-20	1982
		5.2-21	1982
		5.2-22	1982
		5.2-23	1982
		5.2-24	1982
		5.2-25	1982
		5.2-26	1990
		5.2-27	1982
		5.2-28	1987
		5.2-29	1982
		5.2-30	1982
		5.2-31	1987
		5.2-32	1987
		5.2-33	1987
		5.2-34	1982
		5.2-35	1987
		5.2-36	1987
		5.2-37	1988
		5.2-38	1987
		5.2-39	1987
		5.2-40	1987
		5.2-41	1987
		5.2-42	1982
		5.2-43	1987
		5.2-44	1990
		5.2-45	1989

Chapter 5	Containment System	Page	Date
		5.2-46	1989
		5.2-47	1989
		5.2-48	1982
		5.2-49	1988
		5.2-50	1989
		5.2-51	1989
		5.2-52	1987
		5.2-53	1988
		5.2-54	1988
		5.2-55	1989
		5.2-56	1990
		5.2-57	1987
		5.2-58	1987
		5.2-59	1991
		5.2-60	1989
		5.2-61	1988
		5.2-62	1987
		5.2-63	1987
		5.2-64	1987
		5.2-65	1987
		5.2-66	1987
		5.2-67	1987
		5.2-68	1988
		5.2-69	1988
		5.2-70	1987
		5.2-71	1988
		5.2-72	1987
		5.2-73	1987
		5.2-74	1987
		5.2-75	1987
		5.2-76	1987
		5.2-77	1988
		5.2-78	1988
		5.2-79	1990
		5.2-80	1991
		5.2-81	1990
		5.2-82	1990
		5.2-83	1990
		5.2-84	1990
		5.2-85	1990
		5.2-86	1990
		5.2-87	1990
		5.2-88	1990
		5.2-89	1993
		5.2-90	1990
		5.2-91	1990
		5.2-92	1990
		5.2-93	1990
		5.2-94	1990
		5.2-95	1990

VOLUME IV

Chapter 5	Containment System	Page	Date
		5.2-96	1990
		5.2-97	1990
		5.2-98	1990
		5.2-99	1990
		5.2-100	1990
		5.2-101	1990
		5.2-102	1990
		5.2-103	1990
		5.2-104	1990
		5.2-105	1990
		5.2-106	1990
		5.2-107	1990
		5.2-108	1990
		5.2-109	1990
		5.2-110	1990
		5.2-111	1990
		5.2-112	1990
		5.2-113	1991
		5.2-114	1990
		Table 5.2-1	1990
		Table 5.2-2 (2pp)	1990
		Table 5.2-3	1990
		Table 5.2-4	1990
		Table 5.2-5	1990
		Table 5.2-6	1990
		Table 5.2-7	1990
		Fig. 5.2-1	1982
		Fig. 5.2-2	1982
		Fig. 5.2-3	ORIG
		Fig. 5.2-4	1982
		Fig. 5.2-5	1988
		Fig. 5.2.2-1	1982
		Fig. 5.2.2-1A	1982
		Fig. 5.2.2-2	1982
		Fig. 5.2.2-2A	1982
		Fig. 5.2.2-3	1982
		Fig. 5.2.2-4	1982
		Fig. 5.2.2-4A	1982
		Fig. 5.2.2-4B	1982
		Fig. 5.2.2-5	1982
		Fig. 5.2.2-6	1982
		Fig. 5.2.2-6A	1982
		Fig. 5.2.2-6B	1982
		Fig. 5.2.2-6C	1982
		Fig. 5.2.2-6D	1982
		Fig. 5.2.2-7	1982
		Fig. 5.2.2-8	1982
		Fig. 5.2.2-9	1982
		Fig. 5.2.2-10	1982
		Fig. 5.2.2-10A	1982

VOLUME IV

Chapter 5	Containment System	Page	Date
		Fig. 5.2.2-11	1982
		Fig. 5.2.2-11A	1982
		Fig. 5.2.2-12	1982
		Fig. 5.2.2-12A	1982
		Fig. 5.2.2-13	1982
		Fig. 5.2.2-14	1982
		Fig. 5.2.2-15	1982
		Fig. 5.2.2-16	1982
		Fig. 5.2.2-17	1982
		Fig. 5.2.2-18	1982
		Fig. 5.2.2-19	1982
		Fig. 5.2.2-20	1982
		Fig. 5.2.2-21	1982
		Fig. 5.2.2-22	1982
		Fig. 5.2.2-23	1982
		Fig. 5.2.2-24	1982
		Fig. 5.2.2-25	1982
		Fig. 5.2.2-26	1982
		Fig. 5.2.2-27	1982
		Fig. 5.2.2-28	1982
		Fig. 5.2.2-29	1982
		Fig. 5.2.2-30	1982
		Fig. 5.2.2-31	1982
		Fig. 5.2.2-32	1982
		Fig. 5.2.2-33	1982
		Fig. 5.2.2-34	1982
		Fig. 5.2.2-35	1982
		Fig. 5.2.2-36	1982
		Fig. 5.2.2-37	1982
		Fig. 5.2.2-38	1982
		Fig. 5.2.2-39	1982
		Fig. 5.2.2-40	1982
		Fig. 5.2.2-41	1982
		Fig. 5.2.2-42	1982
		Fig. 5.2.2-43	1982
		Fig. 5.2.2-44	1982
		Fig. 5.2.2-45	1982
		Fig. 5.2.2-46	1982
		Fig. 5.2.2-47	1982
		Fig. 5.2.2-48	1982
		Fig. 5.2.2-49	1982
		Fig. 5.2.2-50	1982
		Fig. 5.2.2-51	1982
		Fig. 5.2.2-51A	1982
		Fig. 5.2.2-51B	1982
		Fig. 5.2.2-51C	1982
		Fig. 5.2.2-51D	1982
		Fig. 5.2.2-51E	1982
		Fig. 5.2.2-52	1982
		Fig. 5.2.2-52A	1982
		Fig. 5.2.2-53	1982

VOLUME IV

Chapter 5	Containment System	Page	Date
		Fig. 5.2.2-54	1982
		Fig. 5.2.2-54A	1982
		Fig. 5.2.2-54B	1982
		Fig. 5.2.2-55	1982
		Fig. 5.2.2-55A	1982
		Fig. 5.2.2-56	1982
		Fig. 5.2.2-56A	1982
		Fig. 5.2.2-57	1982
		Fig. 5.2.2-57A	1982
		Fig. 5.2.2-58	1982
		Fig. 5.2.2-58A	1982
		Fig. 5.2.2-59	1982
		Fig. 5.2.2-59A	1982
		Fig. 5.2.2-59B	1982
		Fig. 5.2.2-59C	1982
		Fig. 5.2.2-59D	1982
		Fig. 5.2.2-59E	1982
		Fig. 5.2.2-60	1991
		Fig. 5.2.2-60A	1991
		Fig. 5.2.2-60B	1982
		Fig. 5.2.2-60C	1982
		Fig. 5.2.2-61	1982
		Fig. 5.2.2-62	1982
		Fig. 5.2.2-63	1982
		Fig. 5.2.2-64	1982
		Fig. 5.2.2-65	1982
		Fig. 5.2.2-65A	1982
		5.3-1	1984
		5.3-2	1984
		5.3-3	1982
		5.3-4	1982
		5.3-5	1982
		5.3-6	1982
		5.3-7	1982
		5.3-8	1982
		5.3-9	1982
		5.3-10	1988
		5.3-11	1982
		5.3-12	1993
		5.3-13	1984
		5.3-14	1993
		5.3-15	1982
		5.3-16	1984
		5.3-17	1984
		5.3-18	1982
		5.3-19	1982
		5.3-20	1990
		5.3-21	1982
		5.3-22	1982
		5.3-23	1982

VOLUME IV

Chapter 5	Containment System	Page	Date
		5.3-24	1982
		5.3-25	1982
		5.3-26	1986
		5.3-27	1986
		5.3-28	1982
		5.3-29	1988
		5.3-30	1988
		5.3-31	1988
		5.3-32	1982
		5.3-33	1988
		5.3-34	1989
		Table 5.3-1 (pg 1)	1993
		(pg 2)	1989
		Fig. 5.3-1	1982
		Fig. 5.3-2	1986
		Fig. 5.3-2A	1986
		Fig. 5.3-3	1982
		Fig. 5.3-4	1982
		5.4-1	1992
		5.4-2	1992
		5.4-3	1992
		5.4-4	1992
		5.4-5	1992
		5.4-6	1992
		5.4-7	1992
		5.4-8	1992
		Table 5.4-1 (pg 1)	1993
		(pg 2)	1993
		(pg 3)	1993
		(pg 4)	1993
		(pg 5)	1993
		(pg 6)	1993
		(pg 7)	1993
		(pg 8)	1993
		(pg 9)	1993
		(pg 10)	1993
		(pg 11)	1993
		(pg 12)	1993
		Fig. 5.4-1	1982
		5.5-1	1982
		5.5-2	1982
		5.5-3	1987
		5.5-4	1993
		5.5-5	1993
		5.5-6	1993
		5.5-7	1987
		5.5-8	1993
		5.5-9	1992
		5.5-10	1992
		5.5-11	1992
		5.5-12	1987
		5.5-13	1987

Chapter 5	Containment System	Page	Date
		5.5-14	1987
		5.5-15	1987
		5.5-16	1993
		5.5-17	1993
		Fig. 5.5-1	1985
		Fig. 5.5-2	1985
		Fig. 5.5-3	1982
		5.6-1	1993
		5.6-2	1993
		5.6-3	1992
		5.6-4	1986
		Fig. 5.6-1	1993
		5.7-1	1982
		5.7-2	1982
		5.7-3	1982
		5.7-4	1982
		5.7-5	1982
		5.7-6	1992
		5.7-7	1982
		5.7-8	ORIG
		Fig. 5.7-1	1982
		Fig. 5.7-2	1982

VOLUME IV

Chapter 6	Engineered Safety Features	Page	Date
		6.1-1	1989
		6.1-2	1989
		6.1-3	1989
		6.1-4	1989
		6.1-5	1989
		6.1-6	1989
		6.1-7	1989
		6.1-8	1989
		6.1-9	1989
		6.1-10	1989
		6.1-11	1989
		Table 6.1-1	1989
		6.2-1	1982
		6.2-2	1982
		6.2-3	1982
		6.2-4	1982
		6.2-5	1993
		6.2-6	1993
		6.2-7	1993
		6.2-8	1993
		6.2-9	1988
		6.2-10	1991
		6.2-11	1992
		6.2-12	1992
		6.2-13	1992
		6.2-14	1993
		6.2-15	1982
		6.2-16	1993
		6.2-17	1992
		6.2-18	1992
		6.2-19	1982
		6.2-20	1982
		6.2-21	1990
		6.2-22	1990
		6.2-23	1990
		6.2-24	1990
		6.2-25	1982
		6.2-26	1982
		6.2-27	1982
		6.2-28	1982
		6.2-29	1982
		6.2-30	1993
		6.2-31	1993
		6.2-32	1982
		6.2-33	1993
		6.2-34	1982
		6.2-35	1982
		6.2-36	1982
		6.2-37	1993

VOLUME IV

VOLUME V

Chapter 7	Instrumentation and Control	Page	Date
		7.1-1	1993
		7.1-2	1982
		7.2-1	1982
		7.2-2	1990
		7.2-3	1982
		7.2-4	1982
		7.2-5	1982
		7.2-6	1982
		7.2-7	1982
		7.2-8	1982
		7.2-9	1982
		7.2-10	1982
		7.2-11	1982
		7.2-12	1982
		7.2-13	1990
		7.2-14	1982
		7.2-15	1982
		7.2-16	1982
		7.2-17	1982
		7.2-18	1982
		7.2-19	1982
		7.2-20	1982
		7.2-21	1990
		7.2-22	1988
		7.2-23	1982
		7.2-24	1982
		7.2-25	1986
		7.2-26	1987
		7.2-27	1982
		7.2-28	1982
		7.2-29	1992
		7.2-30	1987
		7.2-31	1992
		7.2-32	1987
		7.2-33	1987
		7.2-34	1987
		7.2-35	1987
		7.2-36	1992
		7.2-37	1987
		7.2-38	1990
		7.2-39	1987
		7.2-40	1987
		7.2-41	1987
		7.2-42	1990
		7.2-43	1987
		7.2-44	1987
		7.2-45	1987
		7.2-46	1990

VOLUME V

Chapter 7 Instrumentation and Control

Page		Date
7.2-47		1987
7.2-48		1987
7.2-49		1987
7.2-50		1987
7.2-51		1987
7.2-52		1987
7.2-53		1987
Table 7.2-1	(pg 1)	1989
	(pg 2)	1992
	(pg 3)	1989
	(pg 4)	1990
	(pg 5)	1990
Table 7.2-2	(pg 1)	1991
	(pg 2)	1992
	(pg 3)	1991
	(pg 4)	1991
	(pg 5)	1991
	(pg 6)	1991
Table 7.2-3		1991
Table 7.2-4		1991
Table 7.2-5	(2pp)	1991
Fig. 7.2-1a		1990
Fig. 7.2-1b		1990
Fig. 7.2-1c		1990
Fig. 7.2-1d		1990
Fig. 7.2-2		1982
Fig. 7.2-3		1982
Fig. 7.2-4		1982
Fig. 7.2-5		1982
Fig. 7.2-6		1982
Fig. 7.2-7		1982
Fig. 7.2-8		1982
Fig. 7.2-9		1982
7.3-1		1987
7.3-2		1990
7.3-3		1982
7.3-4		1982
7.3-5		1982
7.3-6		1992
7.3-7		1983
7.3-8		1982
7.3-9		1982
7.3-10		1990
7.3-11		1982
7.3-12		1982
7.3-13		1982
Fig. 7.3-1		1982
7.4-1		1982
7.4-2		1991

VOLUME V

Chapter 7	Instrumentation and Control	Page	Date
		7.5-1	1982
		7.5-2	1982
		7.5-3	1982
		7.5-4	1982
		7.5-5	1982
		7.5-6	1991
		7.5-7	1987
		7.5-8	1987
		7.5-9	1989
		7.5-10	1987
		7.5-11	1982
		7.5-12	1982
		7.5-13	1991
		7.5-14	1982
		7.5-15	1982
		7.5-16	1982
		7.5-17	1982
		7.5-18	1991
		7.5-19	1982
		7.5-20	1989
		Table 7.5-1	1989
		Table 7.5-2 (2pp)	1989
		Fig. 7.5-1	1982
		Fig. 7.5-2	1982
		Fig. 7.5-3	1982
		7.6-1	1991
		7.6-2	1991
		7.6-3	1991
		7.6-4	1991
		7.6-5	1991
		Fig. 7.6-1	1982
		Fig. 7.6-2	1982
		Fig. 7.6-3	1982
		7.7-1	1982
		7.7-2	1982
		7.7-3	1982
		7.7-4	1982
		7.7-5	1982
		7.7-6	1993
		7.7-7	1991
		7.7-8	1991
		7.7-9	1982
		7.7-10	1986
		7.7-11	1982
		7.8-1	1992
		7.8-2	1992

VOLUME V

VOLUME V

<u>VOLUME V</u>

Chapter 9	Auxiliary and Emergency Systems	Page		Date
		9.1-1		1992
		9.1-2		1982
		9.1-3		1990
		9.1-4		1982
		9.2-1		1982
		9.2-2		1982
		9.2-3		1991
		9.2-4		1986
		9.2-5		1982
		9.2-6		1982
		9.2-7		1988
		9.2-8		1987
		9.2-9		1982
		9.2-10		1982
		9.2-11		1982
		9.2-12		1982
		9.2-13		1982
		9.2-14		1982
		9.2-15		1982
		9.2-16		1990
		9.2-17		1992
		9.2-18		1992
		9.2-19		1982
		9.2-20		1982
		9.2-21		1982
		9.2-22		1982
		9.2-23		1993
		9.2-24		1988
		9.2-25		1982
		9.2-26		1982
		9.2-27		1982
		9.2-28		1982
		9.2-29		1993
		9.2-30		1993
		9.2-31		1982
		9.2-32		1982
		9.2-33		1993
		9.2-34		1991
		9.2-35		1982
		9.2-36		1982
		9.2-37		1982
		9.2-38		1982
		Table 9.2-1		1990
		Table 9.2-2		1989
		Table 9.2-3	(pg 1)	1989
			(pg 2)	1990
			(pg 3)	1989
			(pg 4)	1992
			(pg 5)	1989
			(pg 6)	1990

VOLUME V

Chapter 9	Auxiliary and Emergency Systems	Page	Date
		Table 9.2-3 (pg 7)	1989
		(pg 8)	1989
		(pg 9)	1989
		(pg 10)	1990
		(pg 11)	1989
		(pg 12)	1989
		(pg 13)	1990
		(pg 14)	1990
		(pg 15)	1989
		(pg 16)	1989
		(pg 17)	1989
		Table 9.2-4 (2pp)	1989
		Fig. 9.2-1	1992
		Fig. 9.2-2	1982
		Fig. 9.2-3	1993
		Fig. 9.2-4	1984
		Fig. 9.2-5	ORIG
		Fig. 9.2-6	1982
		9.3-1	1982
		9.3-2	1982
		9.3-3	1990
		9.3-4	1991
		9.3-5	1982
		9.3-6	1982
		9.3-7	1988
		9.3-8	1988
		9.3-9	1987
		9.3-10	1986
		9.3-11	1993
		9.3-12	1990
		9.3-13	1990
		Table 9.3-1	1990
		Table 9.3-2 (1pp)	1990
		Table 9.3-2 (2pp)	1991
		Table 9.3-3 (3pp)	1990
		Fig. 9.3-1	1991
		9.4-1	1982
		9.4-2	1982
		9.4-3	1989
		9.4-4	1983
		9.4-5	1983
		9.4-6	1982
		9.4-7	1986
		Table 9.4-1	1990
		Table 9.4-2 (4pp)	1989
		Table 9.4-3	1989
		Fig. 9.4-1	1990
		9.5-1	1985
		9.5-2	1993
		9.5-3	1992

Chapter 9	Auxiliary and Emergency Systems	Page	Date
		9.5-4	1982
		9.5-5	1982
		9.5-6	1982
		9.5-7	1982
		9.5-8	1982
		9.5-9	1982
		Table 9.5-1	1990
		Table 9.5-2	1993
		Table 9.5-3	1993
		Table 9.5-4	1993
		Fig. 9.5-1	1993
		9.6-1	1991
		9.6-2	1983
		9.6-3	1989
		9.6-4	1987
		9.6-5	1987
		9.6-6	1988
		9.6-7	1983
		9.6-8	1983
		9.6-9	1982
		Fig. 9.6-1	1992
		Fig. 9.6-2	1987
		9.7-1	1982
		9.7-2	1984
		9.7-3	1982
		9.7-4	1982
		9.7-5	1987
		9.7-6	1991
		9.7-7	1991
		9.7-8	1992
		9.7-9	1992
		9.7-10	1991
		9.7-11	1991
		9.7-12	1991
		9.7-13	1991
		9.7-14	1991
		9.7-15	1992
		9.7-16	1991
		9.7-17	1991
		9.7-18	1991
		9.7-19	1991
		9.7-20	1991
		9.7-21	1991
		9.7-22	1991
		9.7-23	1991
		9.7-24	1991
		9.7-25	1991
		Fig. 9.7-1	1982
		Fig. 9.7-2	1982
		Fig. 9.7-3	1991

VOLUME VI

Chapter 10	Steam and Power Conversion System	Page	Date
		10.1-1	1987
		10.1-2	1982
		10.2-1	1982
		10.2-2	1983
		10.2-3	1985
		10.2-4	1991
		10.2-5	1985
		10.2-6	1988
		Fig. 10.2-1	1984
		Fig. 10.2-1A	1991
		Fig. 10.2-1B	1982
		Fig. 10.2-1C	1992
		10.3-1	1986
		10.3-2	1982
		10.3-3	1986
		10.3-4	1984
		10.3-5	1992
		10.3-6	1984
		Fig. 10.3-1	1984
		Fig. 10.3-1A	1984
		10.4-1	1991
		10.4-2	1985
		10.4-3	1991
		10.5-1	1986
		10.5-2	1991
		10.5-3	1982
		10.5-4	1987
		10.5-5	1990
		10.5-6	1989
		10.5-7	1993
		Table 10.5-1	1991
		Fig. 10.5-1	1993
		Fig. 10.5-2	1982
		Fig. 10.5-2A	1982
		Fig. 10.5-3	1982
		Fig. 10.5-3A	1982
		Fig. 10.5-4	1991
		Fig. 10.5-4A	1991
		Fig. 10.5-5	1990
		Fig. 10.5-5A	1982
		10.6-1	1982
		10.6-2	1982
		10.6-3	1982
		10.6-4	1991
		Table 10.6-1	1989
		Fig. 10.6-1	1982

VOLUME VI

Chapter 10	Steam and Power Conversion System	Page	Date
		10.7-1	1982
		10.7-2	1982
		10.8-1	1982
		10.9-1	1988
		10.10-1	1986
		10.11-1	1983
		10.11-2	1988
		10.11-3	1983

VOLUME VI

Chapter 11	Waste Disposal and Radiation Protection System	Page	Date
		11.1-1	1985
		11.1-2	1992
		11.1-3	1982
		11.1-4	1983
		11.1-5	1993
		11.1-6	1985
		11.1-7	1992
		11.1-8	1983
		11.1-9	1992
		11.1-10	1982
		11.1-11	1982
		11.1-12	1993
		11.1-13	1985
		11.1-14	1985
		11.1-15	1985
		11.1-16	1982
		11.1-17	1982
		11.1-18	1992
		Table 11.1-1	1989
		Table 11.1-2	1990
		Table 11.1-3 (pg 1)	1990
		(pg 2)	1993
		Table 11.1-4	1989
		Table 11.1-5	1989
		Table 11.1-6	1989
		Fig. 11.1-1	1988
		Fig. 11.1-2	1990
		Fig. 11.1-2A	1982
		Fig. 11.1-2B	1985
		Fig. 11.1-3	1982
		Fig. 11.1-4	1988
		11.2-1	1983
		11.2-2	1983
		11.2-3	1982
		11.2-4	1982
		11.2-5	1982
		11.2-6	1982
		11.2-7	1982
		11.2-8	1982
		11.2-9	1983
		Table 11.2-1	1989
		Table 11.2-2	1989
		Table 11.2-3	1989
		Table 11.2-4	1989
		Table 11.2-5	1989
		Table 11.2-6	1989
		Table 11.2-7	1989
		Table 11.2-8	1989

VOLUME VI

Chapter 11	Waste Disposal and Radiation Protection System	Page	Date
		Fig. 11.2-1	1982
		11.3-1	1990
		11.3-2	1986
		11.3-3	1988
		11.3-4	1990
		11.3-5	1992
		11.3-6	1992
		11.3-7	1992
		11.3-8	1990
		11.3-9	1990
		11.3-10	1993
		11.3-11	1990
		11.3-12	1992
		11.3-13	1990
		11.3-14	1990
		11.3-15	1992
		11.3-16	1992
		11.3-17	1993
		11.3-18	1993
		Table 11.3-1 (1pp)	1990
		Table 11.3-1 (2pp)	1991
		Table 11.3-1 (3pp)	1991
		Table 11.3-1 (4pp)	1991
		Table 11.3-1 (5pp)	1991
		Table 11.3-2	1990
		11.4-1	1993
		11.4-2	1993
		11.4-3	1993
		11.4-4	1993
		11.4-5	1993
		11.4-6	1993
		11.4-7	1993
		11.4-8	1993
		11.4-9	1993
		11.4-10	1993
		11.4-11	1993
		11.4-12	1991
		11.4-13	1991
		Fig. 11.4-1	1993
		Fig. 11.4-2	1993
		Fig. 11.4-3	1992
		Fig. 11.4-4	1992
		11.5-1	1987
		11.5-2	1990
		11.5-3	1983
		Table 11.5-1	1989
		Table 11.5-2	1989
		Fig. 11.5-1	1983

VOLUME VI

Chapter 11	Waste Disposal and Radiation Protection System	Page	Date
		11.6-1	1988
		11.6-2	1988
		11.6-3	1990
		11.6-4	1993
		Fig. 11.6-1	1990
		Fig. 11.6-2a	1986
		Fig. 11.6-2b	1986
		Fig. 11.6-2c	1990

VOLUME VI

Chapter 12	Conduct of Operations	Page	Date
		12.1-1	1991
		12.1-2	1982
		12.2-1	1991
		12.2-2	1988
		12.2-3	1988
		12.2-4	1988
		12.2-5	1988
		12.2-6	1988
		12.2-7	1988
		12.2-8	1988
		12.3-1	1988
		12.4-1	1983
		12.5-1	1982
		12.6-1	1991
		12.6-2	1991
		12.6-3	ORIG
		12.7-1	1991

Chapter 13 Initial Tests and Operation

VOLUME VII

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		14.1.3-1	1993
		14.1.3-2	1992
		14.1.3-3	1992
		14.1.3-4	1992
		14.1.3-5	1990
		14.1.3-6	1990
		14.1.3-7	1992
		Fig. 14.1.3-1	1990
		Fig. 14.1.3-2	1990
		14.1.4-1	1990
		14.1.5-1	1990
		14.1.5-2	1992
		14.1.5-3	1992
		14.1.5-4	1990
		14.1.6-1	1990
		14.1.6-2	1990
		14.1.6-3	1990
		14.1.6-4	1990
		14.1.6-5	1990
		14.1.6-6	1990
		14.1.6-7	1990
		14.1.6-8	1990
		Fig. 14.1.6-1	1990
		Fig. 14.1.6-2	1990
		Fig. 14.1.6-3	1990
		Fig. 14.1.6-4	1990
		Fig. 14.1.6-5	1990
		Fig. 14.1.6-6	1990
		Fig. 14.1.6-7	1990
		Fig. 14.1.6-8	1990
		Fig. 14.1.6-9	1990
		Fig. 14.1.6-10	1990
		Fig. 14.1.6-11	1990
		Fig. 14.1.6-12	1990
		14.1.7-1	1990
		14.1.7-2	1990
		14.1.7-3	1990
		Fig. 14.1.7-1	1982
		Fig. 14.1.7-2	1982
		14.1.8-1	1990
		14.1.8-2	1990
		14.1.8-3	1990
		14.1.8-4	1990
		14.1.8-5	1990
		Fig. 14.1.8-1	1990
		Fig. 14.1.8-2	1990
		Fig. 14.1.8-3	1990
		Fig. 14.1.8-4	1990
		Fig. 14.1.8-5	1990

VOLUME VII

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		14.1.13-7	1990
		14.1.13-8	1991
		14.1.13-9	1990
		14.1.13-10	1990
		14.1.13-11	1990
		14.1.13-12	1990
		14.1.13-13	1982
		14.1.13-14	1990
		14.1.13-15	1991
		14.1.13-16	1990
		Table 14.1.13-1	1990
		Fig. 14.1.13-1	1982
		Fig. 14.1.13-2	1982
		Fig. 14.1.13-3	1982
		Fig. 14.1.13-4	1982
		Fig. 14.1.13-5	1982
		Fig. 14.1.13-6	1982
		14.2.1-1	1990
		14.2.1-2	1990
		14.2.1-3	1990
		14.2.1-4	1990
		14.2.1-5	1990
		14.2.1-6	1993
		14.2.1-7	1993
		14.2.1-8	1993
		14.2.1-9	1993
		14.2.1-10	1993
		14.2.1-11	1993
		14.2.1-12	1993
		14.2.1-13	1993
		14.2.1-14	1993
		Table 14.2.1-1	1993
		Table 14.2.1-2	1993
		Table 14.2.1-3	1993
		Table 14.2.1-4	1993
		Table 14.2.1-5	1993
		14.2.2-1	1993
		14.2.2-2	1993
		14.2.2-3	1993
		14.2.2-4	1990
		14.2.3-1	1990
		14.2.3-2	1990
		Table 14.2.3-1	1990
		Table 14.2.3-2	1990
		14.2.4-1	1990
		14.2.4-2	1990
		14.2.4-3	1990
		14.2.4-5	1992
		14.2.4-6	1990
		14.2.4-7	1990
		14.2.4-8	1992
		14.2.4-9	1992

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		Fig. 14.2.4-1	1982
		14.2.5-1	1990
		14.2.5-2	1990
		14.2.5-3	1990
		14.2.5-4	1990
		14.2.5-5	1990
		14.2.5-6	1990
		14.2.5-7	1990
		14.2.5-8	1990
		14.2.5-9	1990
		Table 14.2.5-1	1990
		Fig. 14.2.5-1	1990
		Fig. 14.2.5-2	1990
		Fig. 14.2.5-3	1990
		Fig. 14.2.5-4	1990
		Fig. 14.2.5-5	1990
		Fig. 14.2.5-6	1990
		14.2.6-1	1990
		14.2.6-2	1990
		14.2.6-3	1990
		14.2.6-4	1990
		14.2.6-5	1990
		14.2.6-6	1990
		14.2.6-7	1990
		14.2.6-8	1990
		14.2.6-9	1990
		14.2.6-10	1990
		14.2.6-11	1990
		14.2.6-12	1990
		14.2.6-13	1990
		14.2.6-14	1990
		14.2.6-15	1990
		Table 14.2.6-1	1990
		Fig. 14.2.6-1	1990
		Fig. 14.2.6-2	1990
		Fig. 14.2.6-3	1990
		Fig. 14.2.6-4	1990
		14.2.7-1	1990
		14.2.7-2	1990
		14.2.7-3	1990
		14.2.7-4	1990
		14.2.7-5	1990
		14.2.7-6	1990
		Table 14.2.7-1	1990
		Table 14.2.7-2	1990
		Table 14.2.7-3	1992
		Fig. 14.2.7-1	1982
		Fig. 14.2.7-2	1982
		Fig. 14.2.7-3	1987

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		Fig. 14.2.7-4	1987
		Fig. 14.2.7-5	1982
		Fig. 14.2.7-6	1982
		Fig. 14.2.7-7	1982
		Fig. 14.2.7-8	1982
		Fig. 14.2.7-9	1990
		Fig. 14.2.7-10	1990
		Fig. 14.2.7-11	1990
		Fig. 14.2.7-12	1990
		14.2.8-1	1990
		14.2.8-2	1990
		14.2.8-3	1990
		14.2.8-4	1990
		14.2.8-5	1990
		14.2.8-6	1990
		14.2.8-7	1990
		Table 14.2.8-1	1990
		Fig. 14.2.8-1	1990
		Fig. 14.2.8-2	1990
		Fig. 14.2.8-3	1990
		Fig. 14.2.8-4	1990
		Fig. 14.2.8-5	1990
		Fig. 14.2.8-6	1990
		Fig. 14.2.8-7	1990
		14.3.1-1	1993
		14.3.1-2	1993
		14.3.1-3	1993
		14.3.1-4	1993
		14.3.1-5	1993
		14.3.1-6	1993
		14.3.1-7	1993
		14.3.1-8	1993
		14.3.1-9	1993
		14.3.1-10	1990
		14.3.1-11	1992
		14.3.1-11a	1993
		14.3.1-12	1993
		14.3.1-13	1993
		Table 14.3.1-1 (pg 1)	1993
		Table 14.3.1-1 (pg 2)	1992
		Table 14.3.1-1 (pg 3)	1990
		Table 14.3.1-2	1990
		Table 14.3.1-3 (2pp)	1990
		Table 14.3.1-4	1990
		Table 14.3.1-5	1990
		Table 14.3.1-6	1990
		Fig. 14.3.1-1a	1990
		Fig. 14.3.1-1b	1990
		Fig. 14.3.1-1c	1990

VOLUME VII

VOLUME VII

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		Fig. 14.3.1-8d	1990
		Fig. 14.3.1-8e	1990
		Fig. 14.3.1-8f	1990
		Fig. 14.3.1-8g	1990
		Fig. 14.3.1-9a	1990
		Fig. 14.3.1-9b	1990
		Fig. 14.3.1-9c	1990
		Fig. 14.3.1-9d	1990
		Fig. 14.3.1-9e	1990
		Fig. 14.3.1-9f	1990
		Fig. 14.3.1-9g	1990
		Fig. 14.3.1-10a	1990
		Fig. 14.3.1-10b	1990
		Fig. 14.3.1-10c	1990
		Fig. 14.3.1-10d	1990
		Fig. 14.3.1-10e	1990
		Fig. 14.3.1-10f	1990
		Fig. 14.3.1-10g	1990
		Fig. 14.3.1-11a	1990
		Fig. 14.3.1-11b	1990
		Fig. 14.3.1-11c	1990
		Fig. 14.3.1-11d	1990
		Fig. 14.3.1-11e	1990
		Fig. 14.3.1-11f	1990
		Fig. 14.3.1-11g	1990
		Fig. 14.3.1-12a	1990
		Fig. 14.3.1-12b	1990
		Fig. 14.3.1-12c	1990
		Fig. 14.3.1-12d	1990
		Fig. 14.3.1-12e	1990
		Fig. 14.3.1-12f	1990
		Fig. 14.3.1-12g	1990
		Fig. 14.3.1-13a	1990
		Fig. 14.3.1-13b	1990
		Fig. 14.3.1-13c	1990
		Fig. 14.3.1-13d	1990
		Fig. 14.3.1-13e	1990
		Fig. 14.3.1-13f	1990
		Fig. 14.3.1-13g	1990
		Fig. 14.3.1-14	1990
		Fig. 14.3.1-15	1990
		Fig. 14.3.1-16	1990
		Fig. 14.3.1-17	1990
		Fig. 14.3.1-18	1990
		Fig. 14.3.1-19	1990
		Fig. 14.3.1-20	1990
		Fig. 14.3.1-21	1990
		Fig. 14.3.1-22	1990
		Fig. 14.3.1-23	1990

VOLUME VII

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		14.3.2-1	1990
		14.3.2-2	1990
		14.3.2-3	1990
		14.3.2-4	1990
		14.3.2-5	1990
		14.3.2-6	1993
		14.3.2-7	1993
		Table 14.3.2-1	1990
		Table 14.3.2-2	1990
		Table 14.3.2-3	1993
		Table 14.3.2-4	1990
		Table 14.3.2-5	1990
		Table 14.3.2-6	1990
		Table 14.3.2-7	1993
		Fig. 14.3.2-1	1990
		Fig. 14.3.2-2	1990
		Fig. 14.3.2-3	1990
		Fig. 14.3.2-4	1990
		Fig. 14.3.2-5	1990
		Fig. 14.3.2-6	1990
		Fig. 14.3.2-7	1990
		Fig. 14.3.2-8	1990
		Fig. 14.3.2-9	1990
		Fig. 14.3.2-10	1990
		Fig. 14.3.2-11	1990
		Fig. 14.3.2-12	1990
		Fig. 14.3.2-13	1990
		Fig. 14.3.2-14	1990
		Fig. 14.3.2-15	1990
		Fig. 14.3.2-16	1990
		Fig. 14.3.2-17	1990
		Fig. 14.3.2-18	1990
		Fig. 14.3.2-19	1990
		Fig. 14.3.2-20	1990
		Fig. 14.3.2-21	1990
		Fig. 14.3.2-22	1990
		Fig. 14.3.2-23	1990
		Fig. 14.3.2-24	1990
		Fig. 14.3.2-25	1990
		Fig. 14.3.2-26	1990
		Fig. 14.3.2-27	1990
		Fig. 14.3.2-28	1990
		Fig. 14.3.2-29	1990
		Fig. 14.3.2-30	1990
		Fig. 14.3.2-31	1990
		Fig. 14.3.2-32	1990
		Fig. 14.3.2-33	1990
		Fig. 14.3.2-34	1990
		Fig. 14.3.2-35	1990

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		Fig. 14.3.2-36	1990
		Fig. 14.3.2-37	1990
		Fig. 14.3.2-38	1990
		Fig. 14.3.2-39	1990
		Fig. 14.3.2-40	1990
		14.3.3-1	1989
		14.3.3-2	1988
		14.3.4-1	1992
		14.3.4-2	1992
		14.3.4-3	1992
		14.3.4-4	1992
		14.3.4-5	1992
		14.3.4-6	1992
		14.3.4-7	1992
		14.3.4-8	1992
		14.3.4-9	1992
		14.3.4-10	1992
		14.3.4-11	1992
		14.3.4-12	1992
		14.3.4-13	1992
		14.3.4-14	1992
		14.3.4-15	1992
		14.3.4-16	1992
		14.3.4-17	1992
		14.3.4-18	1992
		14.3.4-19	1992
		14.3.4-20	1992
		14.3.4-21	1992
		14.3.4-22	1992
		14.3.4-23	1992
		14.3.4-24	1992
		14.3.4-25	1992
		14.3.4-26	1992
		14.3.4-27	1992
		14.3.4-28	1992
		14.3.4-29	1992
		14.3.4-30	1992
		14.3.4-31	1992
		14.3.4-32	1992
		14.3.4-33	1992
		14.3.4-34	1992
		14.3.4-35	1992
		14.3.4-36	1992
		14.3.4-37	1992
		14.3.4-38	1992
		14.3.4-39	1992
		14.3.4-40	1992
		14.3.4-41	1992
		14.3.4-42	1992

Chapter 14 Safety Analysis (Unit 1)

VOLUME VIII

Chapter 14 — Safety Analysis (Unit 1)

Page	Date
Table 14.3.4-6	1992
Table 14.3.4-7 (2pp)	1992
Table 14.3.2-8 (2pp)	1992
Table 14.3.4-9	1992
Table 14.3.4-10	1992
Table 14.3.4-11	1992
Table 14.3.4-12	1992
Table 14.3.4-13	1992
Table 14.3.4-14	1992
Table 14.3.4-15 (2 pp)	1992
Table 14.3.4-16 (2 pp)	1992
Table 14.3.4-17	1992
Table 14.3-4-18	1992
Table 14.3.4-19	1992
Table 14.3.4-20	1992
Table 14.3.4-21 (2pp)	1992
Table 14.3.4-22	1992
Table 14.3.4-23	1992
Table 14.3.4-24	1992
Table 14.3.4-25 (3pp)	1992
Table 14.3.4-26	1992
Table 14.3.4-27 (10pp)	1992
Table 14.3.4-28	1992
Table 14.3.4-29 (2pp)	1992
Table 14.3.4-30	1992
Table 14.3.4-31 (2pp)	1992
Table 14.3.4-32 (2pp)	1992
Table 14.3.4-33 (2pp)	1992
Table 14.3.4-34 (2pp)	1992
Table 14.3.4-35 (2pp)	1992
Table 14.3.4-36 (2pp)	1992
Table 14.3.4-37	1992
Table 14.3.4-38	1992
Table 14.3.4-39 (2pp)	1992
Table 14.3.4-40 (2pp)	1992
Table 14.3.4-41	1992
Table 14.3.4-42	1992
Table 14.3.4-43	1992
Table 14.3.4-44	1992
Table 14.3.4-45	1992
Table 14.3.4-46 (2pp)	
Fig. 14.3.4-1	1992
Fig. 14.3.4-2	1992
Fig. 14.3.4-3	1992
Fig. 14.3.4-4	1992
Fig. 14.3.4-5	1992
Fig. 14.3.4-6	1992
Fig. 14.3.4-7	1992

VOLUME VIII

VOLUME IX

VOLUME IX

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		Table 14.3.5-9 pg 1	1990
		pg 2	1990
		pg 3	1990
		pg 4	1990
		pg 5	1990
		pg 6	1990
		Fig. 14.3.5-1	1982
		Fig. 14.3.5-2	1982
		Fig. 14.3.5-3	1982
		Fig. 14.3.5-4	1982
		Fig. 14.3.5-5	1987
		Fig. 14.3.5-6	1987
		14.3.6-1	1989
		14.3.6-2	1982
		14.3.6-3	1991
		14.3.6-4	1989
		14.3.6-5	1982
		14.3.6-6	1982
		14.3.6-7	1982
		14.3.6-8	1982
		14.3.6-9	1982
		14.3.6-10	1982
		14.3.6-11	1990
		14.3.6-12	1990
		14.3.6-13	1990
		14.3.6-14	1990
		14.3.6-15	1990
		14.3.6-16	1990
		14.3.6-17	1990
		14.3.6-18	1990
		14.3.6-19	1990
		14.3.6-20	1990
		14.3.6-21	1990
		14.3.6-22	1990
		14.3.6-23	1990
		14.3.6-24	1990
		14.3.6-25	1990
		14.3.6-26	1990
		14.3.6-27	1991
		14.3.6-28	1990
		14.3.6-29	1991
		Table 14.3.6-1	1990
		Table 14.3.6-2	1990
		Table 14.3.6-3	1990
		Table 14.3.6-4	1990
		Table 14.3.6-5	1990
		Table 14.3.6-6	1990
		Table 14.3.6-7	1990
		Table 14.3.6-8	1990

VOLUME IX

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		Table 14.3.6-9	1990
		Table 14.3.6-10	1990
		Table 14.3.6-11	1990
		Table 14.3.6-12	1990
		Table 14.3.6-13	1990
		Table 14.3.6-14	1990
		Table 14.3.6-15	1990
		Table 14.3.6-16	1990
		Table 14.3.6-17 (2pp)	1990
		Fig. 14.3.6-1	1982
		Fig. 14.3.6-2	1982
		Fig. 14.3.6-3	DELETED
		Fig. 14.3.6-4	DELETED
		Fig. 14.3.6-5	DELETED
		Fig. 14.3.6-6	1990
		Fig. 14.3.6-7	1990
		Fig. 14.3.6-8	1990
		Fig. 14.3.6-9	1990
		Fig. 14.3.6-10	1990
		Fig. 14.3.6-11	1990
		Fig. 14.3.6-12	1990
		Fig. 14.3.6-13	1990
		Fig. 14.3.6-14	1990
		Fig. 14.3.6-14A	1990
		Fig. 14.3.6-15	1990
		Fig. 14.3.6-16	1990
		Fig. 14.3.6-17	1990
		Fig. 14.3.6-18	1990
		Fig. 14.3.6-19	1990
		Fig. 14.3.6-20	1990
		Fig. 14.3.6-21	1990
		Fig. 14.3.6-22	1990
		14.3.7-1	1993
		14.3.8-1	1993
		14.4.1-1	1990
		14.4.2-1	1987
		14.4.2-2	1982
		14.4.2-3	1982
		14.4.2-4	1982
		14.4.2-5	1982
		14.4.2-6	1982
		14.4.2-7	1982
		14.4.2-8	1982
		14.4.2-9	1982
		14.4.2-10	1982
		14.4.2-11	1982
		14.4.2-12	1982
		14.4.2-13	1987
		14.4.2-14	1982

VOLUME IX

Chapter 14	Safety Analysis (Unit 1)	Page		Date
		Fig. 14.4.6-10		1993
		Fig. 14.4.6-10a		1993
		Fig. 14.4.6-11		1993
		Fig. 14.4.6-11a		1992
		14.4.7-1		1990
		14.4.7-2		1990
		14.4.8-1		1990
		14.4.9-1		1990
		14.4.9-2		1990
		14.4.9-3		1990
		Fig. 14.4.9-1		1982
		Fig. 14.4.9-2		1982
		14.4.10-1		1990
		14.4.10-2		1990
		14.4.10-3		1990
		14.4.10-4		1990
		Table 14.4.10-1		1990
		14.4.11-1		1990
		14.4.11-2		1990
		14.4.11-3		1990
		14.4.11-4		1990
		14.4.11-5		1990
		14.4.11-6		1990
		14.4.11-7		1990
		Table 14.4.11-1	(pg 1)	1993
			(pg 2)	1993
			(pg 3)	1990
			(pg 4)	1993
			(pg 5)	1993
			(pg 6)	1992
			(pg 7)	1993
			(pg 8)	1990
			(pg 9)	1993
			(pg 10)	1993
			(pg 11)	1993
		Table 14.4.11-2		1990
		Table 14.4.11-3		1990
		Table 14.4.11-4		1990
		Table 14.4.11-5		1990
		Table 14.4.11-6		1990
		Table 14.4.11-7		1990
		Table 14.4.11-8		1990
		Table 14.4.11-9		1992
		14A-1		1992
		14A-2		1982
		14A-3		1982
		14A-4		1982
		14A-5		1982
		14A-6		1982
		14A-7		1982
		14A-8		1982
		14A-9		1982
		14A-10		1982

VOLUME IX

Chapter 14	Safety Analysis (Unit 1)	Page	Date
		14A-11	1982
		14A-12	1992
		14A-13	1992
		14A-14	1992
		14A-15	1992
		14A-16	1993
		14A-17	1993
		14A-18	1992
		14A-19	1992
		14A-20	1992
		14A-21	1992
		14A-22	1992
		14A-23	1992
		14A-24	1992
		14A-25	1992
		14A-26	1992
		14.G-1	1987
		14.G-2	1987
		14.G-3	1987
		14.G-4	1987
		14.G-5	1987
		14.G-6	1987
		14.G-7	1987
		14.G-8	1987
		14.G-9	1987
		14.G-10	1987
		14.G-11	1987
		14.G-12	1988
		Table 14.G-1	1987
		Table 14.G-2	1987
		Table 14.G-3	1987
		Fig. 14.G-1 (3pp)	1988
		Fig. 14.G-1 Notes	1988
		Fig. 14.G-2	1987

VOLUME X

VOLUME X

Chapter 14	Safety Analysis (Unit 2)	Page	Date
		Fig. 14.1.1-2	1992
		14.1.2A-1	1992
		14.1.2A-2	1992
		14.1.2A-3	1991
		14.1.2A-4	1991
		14.1.2A-5	1991
		14.1.2A-6	1991
		14.1.2A-7	1991
		14.1.2A-8	1991
		Fig. 14.1.2A-1	1991
		Fig. 14.1.2A-2	1991
		Fig. 14.1.2A-3	1991
		Fig. 14.1.2A-4	1991
		Fig. 14.1.2A-5	1991
		Fig. 14.1.2A-6	1991
		Fig. 14.1.2A-7	1991
		Fig. 14.1.2A-8	1991
		Fig. 14.1.2A-9	1991
		14.1.2B-1	1991
		14.1.2B-2	1993
		14.1.2B-3	1991
		14.1.2B-4	1991
		14.1.2B-5	1991
		14.1.2B-6	1991
		14.1.2B-7	1991
		Fig. 14.1.2B-1	1991
		Fig. 14.1.2B-2	1991
		Fig. 14.1.2B-3	1991
		Fig. 14.1.2B-4	1991
		Fig. 14.1.2B-5	1991
		Fig. 14.1.2B-6	1991
		Fig. 14.1.2B-7	1991
		Fig. 14.1.2B-8	1991
		Fig. 14.1.2B-9	1991
		14.1.3-1	1993
		14.1.3-2	1991
		14.1.3-3	1991
		14.1.3-4	1993
		14.1.3-5	1991
		14.1.3-6	1991
		14.1.3-7	1991
		14.1.3.4 Refs.	1991
		Fig. 14.1.3-1	1991
		Fig. 14.1.3-2	1991
		14.1.4-1	1991
		14.1.5-1	1991
		14.1.5-2	1991
		14.1.5-3	1991
		14.1.5-4	1991
		14.1.5-5	1991

VOLUME X

Chapter 14 Safety Analysis (Unit 2)

	Page	Date
	14.1.5-6	1991
	14.1.5-7	1991
	14.1.6-1	1991
	14.1.6-2	1991
	14.1.6-3	1991
	14.1.6-4	1991
	14.1.6-5	1991
	14.1.6-6	1991
	14.1.6-7	1991
	14.1.6-8	1991
	14.1.6-9	1991
	14.1.6-10	1991
	14.1.6-11	1991
Table	14.1.6-1	1991
Table	14.1.6-2	1991
Fig.	14.1.6-1	1991
Fig.	14.1.6-2	1991
Fig.	14.1.6-3	1991
Fig.	14.1.6-4	1991
Fig.	14.1.6-5	1991
Fig.	14.1.6-6	1991
Fig.	14.1.6-7	1991
Fig.	14.1.6-8	1991
Fig.	14.1.6-9	1991
Fig.	14.1.6-10	1991
Fig.	14.1.6-11	1991
Fig.	14.1.6-12	1991
	14.1.7-1	1991
	14.1.7-2	1991
	14.1.7-3	1991
	14.1.7-4	1991
Table	14.1.7-5	1991
Fig.	14.1.7-1	1992
	14.1.8A-1	1991
	14.1.8A-2	1991
	14.1.8A-3	1991
	14.1.8A-4	1991
	14.1.8A-5	1991
	14.1.8A-6	1991
Table	14.1.8A-1 (2pp)	1991
Fig.	14.1.8A-1	1991
Fig.	14.1.8A-2	1991
Fig.	14.1.8A-3	1991
Fig.	14.1.8A-4	1991
Fig.	14.1.8A-5	1991
Fig.	14.1.8A-6	1991
Fig.	14.1.8A-7	1991
Fig.	14.1.8A-8	1991
Fig.	14.1.8A-9	1991

VOLUME X

Chapter 14	Safety Analysis (Unit 2)	Page	Date
		Fig. 14.1.8A-10	1991
		Fig. 14.1.8A-11	1991
		Fig. 14.1.8A-12	1991
		14.1.8B-1	1991
		14.1.8B-2	1991
		14.1.8B-3	1991
		14.1.8B-4	1991
		14.1.8B-5	1993
		14.1.8B-6	1991
		14.1.8B-7	1991
		14.1.8B-7	1991
		14.1.8B-8	1991
		Fig. 14.1.8B-1	1991
		Fig. 14.1.8B-2	1991
		Fig. 14.1.8B-3	1991
		Fig. 14.1.8B-4	1991
		Fig. 14.1.8B-5	1991
		Fig. 14.1.8B-6	1991
		Fig. 14.1.8B-7	1991
		Fig. 14.1.8B-8	1991
		Fig. 14.1.8B-9	1991
		Fig. 14.1.8B-10	1991
		Fig. 14.1.8B-11	1991
		Fig. 14.1.8B-12	1991
		14.1.9-1	1991
		14.1.9-2	1991
		14.1.9-3	1991
		14.1.9-4	1991
		14.1.9-5	1991
		14.1.9-6	1991
		Table 14.1.9-1	1991
		Fig. 14.1.9-1	1992
		Fig. 14.1.9-2	1992
		Fig. 14.1.9-3	1992
		14.1.10A-1	1991
		14.1.10A-2	1991
		14.1.10A-3	1993
		14.1.10A-4	1993
		14.1.10A-5	1993
		14.1.10A-6	1993
		14.1.10A-7	1993
		14.1.10A-8	1993
		14.1.10A-9	1991
		Table 14.1.10A-1	1993
		Table 14.1.10A-2	1993
		Table 14.1.10A-3	1993
		Table 14.1.10A-4	1993
		Fig. 14.1.10A-1	1993
		Fig. 14.1.10A-2	1993

Chapter 14	Safety Analysis (Unit 2)	Page	Date
		Fig. 14.1.10A-3	1993
		Fig. 14.1.10A-4	1993
		Fig. 14.1.10A-5	1993
		Fig. 14.1.10A-6	1993
		Fig. 14.1.10A-7	1993
		Fig. 14.1.10A-8	1993
		14.1.10B-1	1991
		14.1.10B-2	1991
		14.1.10B-3	1993
		14.1.10B-4	1993
		14.1.10B-5	1991
		14.1.10B-6	1993
		14.1.10B-7	1993
		14.1.10B-8	1991
		14.1.10B-9	1991
		Table 14.1.10B-1	1993
		Table 14.1.10B-2	1993
		Table 14.1.10B-3	1993
		Table 14.1.10B-4	1993
		Fig. 14.1.10B-1	1993
		Fig. 14.1.10B-2	1993
		Fig. 14.1.10B-3	1993
		Fig. 14.1.10B-4	1993
		Fig. 14.1.10B-5	1993
		Fig. 14.1.10B-6	1993
		Fig. 14.1.10B-7	1993
		Fig. 14.1.10B-8	1993
		14.1.11A-1	1991
		14.1.11A-2	1993
		14.1.11A-3	1991
		14.1.11A-4	1993
		14.1.11A-5	1991
		14.1.11A-6	1991
		Table 14.1.11A-1	1991
		Fig. 14.1.11A-1	1991
		Fig. 14.1.11A-2	1991
		Fig. 14.1.11A-3	1991
		Fig. 14.1.11A-4	1991
		Fig. 14.1.11A-5	1991
		Fig. 14.1.11A-6	1991
		Fig. 14.1.11A-7	1991
		Fig. 14.1.11A-8	1991
		14.1.11B-1	1991
		14.1.11B-2	1993
		14.1.11B-3	1991
		14.1.11B-4	1991
		14.1.11B-5	1991
		Table 14.1.11B-1	1991
		Fig. 14.1.11B-1	1991
		Fig. 14.1.11B-2	1991

VOLUME X

Chapter 14	Safety Analysis (Unit 2)	Page	Date
		Fig. 14.1.11B-3	1991
		Fig. 14.1.11B-4	1991
		Fig. 14.1.11B-5	1991
		Fig. 14.1.11B-6	1991
		Fig. 14.1.11B-7	1991
		Fig. 14.1.11B-8	1991
		14.1.12-1	1991
		14.1.12-2	1991
		14.1.12-3	1991
		14.1.12-4	1991
		14.1.12-5	1991
		Table 14.1.12-1	1991
		Fig. 14.1.12-1	1992
		Fig. 14.1.12-2	1992
		14.1.13-1	1991
		14.2-1	1993
		14.2.1-1	1993
		14.2.2-1	1993
		14.2.2-2	1993
		14.2.2-3	1993
		14.2.2-4	1991
		14.2.2-5	1991
		Table 14.2.2-1	1991
		Table 14.2.2-2	1993
		Table 14.2.2-3	1993
		14.2.3-1	1991
		14.2.4-1	1991
		14.2.4-2	1991
		14.2.4-3	1991
		14.2.4-4	1991
		14.2.4-5	1991
		14.2.4-6	1991
		14.2.4-7	1991
		14.2.4-8	1991
		14.2.4-9	1991
		Table 14.2.4-1	1991
		14.2.5-1	1991
		14.2.5-2	1991
		14.2.5-3	1991
		14.2.5-4	1991
		14.2.5-5	1991
		14.2.5-6	1991
		14.2.5-7	1991
		14.2.5-8	1991
		14.2.5-9	1991
		14.2.5-10	1991
		14.2.5-11	1991
		Table 14.2.5-1	1991
		Table 14.2.5-2 (2pp)	1991
		Fig. 14.2.5-1	1991

VOLUME X

Chapter 14	Safety Analysis (Unit 2)		Page	Date
		Fig.	14.2.5-2	1991
		Fig.	14.2.5-3	1992
		Fig.	14.2.5-4	1992
		Fig.	14.2.5-5	1991
		Fig.	14.2.5-6	1991
			14.2.6-1	1991
			14.2.6-2	1991
			14.2.6-3	1991
			14.2.6-4	1991
			14.2.6-5	1991
			14.2.6-6	1991
			14.2.6-7	1991
			14.2.6-8	1991
			14.2.6-9	1991
			14.2.6-10	1991
			14.2.6-11	1991
			14.2.6-12	1991
			14.2.6-13	1991
			14.2.6-14	1991
			14.2.6-15	1991
			14.2.6-16	1991
			14.2.6-17	1991
		Table	14.2.6-1	1991
		Fig.	14.2.6-1	1991
		Fig.	14.2.6-2	1991
			14.2.7-1	1993
			14.2.8-1	1991
			14.2.8-2	1991
			14.2.8-3	1991
			14.2.8-4	1991
			14.2.8-5	1991
			14.2.8-6	1991
		Table	14.2.8-1 (3pp)	1991
		Fig.	14.2.8-1	1991
		Fig.	14.2.8-2	1991
		Fig.	14.2.8-3	1991
		Fig.	14.2.8-4	1991
		Fig.	14.2.8-5	1991
		Fig.	14.2.8-6	1991
		Fig.	14.2.8-7	1991
		Fig.	14.2.8-8	1991

VOLUME X

VOLUME XI

Chapter 14	Safety Analysis (Unit 2)	Page	Date
		Fig. 14.3.1-22	1991
		Fig. 14.3.1-23	1991
		Fig. 14.3.1-24	1991
		Fig. 14.3.1-25	1991
		Fig. 14.3.1-26	1991
		Fig. 14.3.1-27	1991
		Fig. 14.3.1-28	1991
		Fig. 14.3.1-29	1991
		Fig. 14.3.1-30	1991
		14.3.1-24	1991
		14.3.1-25	1991
		14.3.1-26	1991
		14.3.1-27	1991
		14.3.1-28	1991
		14.3.1-29	1991
		14.3.1-30	1992
		14.3.1-31	1992
		14.3.1-32	1992
		14.3.1-33	1992
		14.3.1-34	1992
		14.3.1-35	1992
		14.3.1-36	1992
		14.3.1-37	1992
		14.3.1-38	1992
		14.3.1-39	1992
		14.3.1-40	1992
		14.3.1-41	1992
		14.3.1-42	1992
		Table 14.3.1-11 (2pp)	1992
		Table 14.3.1-12	1992
		Table 14.3.1-13	1992
		Table 14.3.1-14	1992
		Table 14.3.1-15	1992
		Table 14.3.1-16 (2pp)	1992
		Fig. 14.3.1-31	1991
		Fig. 14.3.1-32	1991
		Fig. 14.3.1-33	1991
		Fig. 14.3.1-34	1991
		Fig. 14.3.1-35	1991
		Fig. 14.3.1-36	1991
		Fig. 14.3.1-37	1991
		Fig. 14.3.1-38	1991
		Fig. 14.3.1-39	1991
		Fig. 14.3.1-40	1991
		Fig. 14.3.1-41	1991
		Fig. 14.3.1-42	1991
		Fig. 14.3.1-43	1991
		Fig. 14.3.1-44	1991
		Fig. 14.3.1-45	1991

VOLUME XI

VOLUME XI

Chapter 14 Safety Analysis (Unit 2)

<u>VOLUME XI</u>

Chapter 14 . Safety Analysis (Unit 2)	Page	Date
	Fig. 14.3.2-37	1991
	Fig. 14.3.2-38	1991
	Fig. 14.3.2-39	1991
	Fig. 14.3.2-40	1991
	Fig. 14.3.2-41	1991
	Fig. 14.3.2-42	1991
	Fig. 14.3.2-43	1991
	Fig. 14.3.2-44	1991
	Fig. 14.3.2-45	1991
	Fig. 14.3.2-46	1991
	Fig. 14.3.2-47	1991
	Fig. 14.3.2-48	1991
	Fig. 14.3.2-49	1991
	Fig. 14.3.2-50	1991
	Fig. 14.3.2-51	1991
	Fig. 14.3.2-52	1991
	Fig. 14.3.2-53	1991
	Fig. 14.3.2-54	1991
	Fig. 14.3.2-55	1991
	Fig. 14.3.2-56	1991
	Fig. 14.3.2-57	1991
	Fig. 14.3.2-58	1991
	Fig. 14.3.2-59	1991
	14.3.3-1	1987
	14.3.3-2	1987
	14.3.3-3	1987
	14.3.3-4	1987
	14.3.3-5	1987
	14.3.3-6	1987
	14.3.4-1	1992
	14.3.5-1	1993
	14.3.5-2	1993
	14.3.5-3	1993
	14.3.5-4	1993
	Table 14.3.5-1	1993
	Table 14.3.5-2	1993
	Table 14.3.5-3	1993
	Table 14.3.5-4	1993
	Table 14.3.5-5 (3 pp)	1993
	Table 14.3.5-6	1993
	14.3.6-1	1990
	14.3.7-1	1993
	14.3.7-2	1993

Chapter 14	Safety Analysis	Page	Date
Appendix J			ORIG

VOLUME XIII

Chapter 14	Safety Analysis	Page	Date
Appendix M			ORIG

TABLE OF CONTENTS (Cont'd)

CHAPTER 1

LIST OF TABLES

CHAPTER 2

TABLE OF CONTENTS (Cont'd)

CHAPTER 2

TABLE OF CONTENTS (Cont'd)

CHAPTER 2

LIST OF TABLES (Cont'd)

CHAPTER 2

LIST OF FIGURES

LIST OF FIGURES (Cont'd.)

LIST OF FIGURES (Cont'd.)

CHAPTER 5

LIST OF TABLES

CHAPTER 5

LIST OF FIGURES

CHAPTER 7

LIST OF TABLES

CHAPTER 7

LIST OF FIGURES

CHAPTER 8

TABLE OF CONTENTS

CHAPTER 8

LIST OF FIGURES

CHAPTER 9

TABLE OF CONTENTS

CHAPTER 9

TABLE OF CONTENTS (Cont'd)

CHAPTER 9

LIST OF TABLES

CHAPTER 9

LIST OF FIGURES

CHAPTER 11

LIST OF TABLES

CHAPTER 11

LIST OF FIGURES

CHAPTER 12

TABLE OF CONTENTS

CHAPTER 12

LIST OF FIGURES

CHAPTER 14

LIST OF TABLES

CHAPTER 14

LIST OF TABLES

CHAPTER 14

LIST OF TABLES

LIST OF TABLES

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

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 1 INTRODUCTION AND SUMMARY

1.0 INTRODUCTION

This Updated Final Safety Analysis Report is submitted in accordance with the requirements of 10 CFR 50.71 (e). It is based on the original FSAR, including 84 amendments, which was submitted in support of an application by Indiana & Michigan Electric Company (I&M), whose name is now Indiana Michigan Power Company (the acronym I&M is still used however) for licenses to operate two nuclear power units at its Donald C. Cook Nuclear Plant.

This submittal contains update information for the period up to six months prior to the most recent revision of this document. The update information is of a similar level of detail as that presented in the original FSAR. It includes changes necessary to reflect information and analysis submitted to the NRC or prepared pursuant to Commission requirements, and it includes changes describing physical modifications to the plant.

I&M and Westinghouse Electric Corporation have jointly participated in the design and construction of each unit. The plant is operated by I&M. Each unit employs a pressurized water reactor nuclear steam supply system furnished by Westinghouse Electric Corporation which is similar in design concept to the majority of the nuclear power plants licensed by the Nuclear Regulatory Commission. Certain components of the auxiliary systems are shared between the two units, but in no case does such sharing result in compromising or impairing the safe and continued operation of either unit. Those systems and components which are shared are identified herein and the effects of the sharing analyzed.

The Unit 1 reactor is currently designed for a power output of 3250 MWt and the Unit 2 reactor is designed for a power output of 3411 MWt, which are their licensed ratings. The approximate gross and net electrical outputs of Unit 1 are 1066 MWe and 1030 MWe and of Unit 2 are 1138 MWe and 1100 MWe, respectively. Containment and engineered safeguards are designed and evaluated for operation at the power rating of 3411 MWt. Most postulated accidents having off-site dose consequences are analyzed at the power rating of 3411 MWt.

The remainder of Chapter 1 of this report summarizes the principal design features and safety criteria of the nuclear units, pointing out the similarities and differences with respect to other pressurized water nuclear power plants employing the same technology and basic engineering features as the Cook Nuclear Plant.

The research and development program is discussed in Section 1.6. The quality assurance program is discussed in Section 1.7.

Chapter 2 contains a description and evaluation of the site and environs, supporting the suitability of that site for a nuclear plant of the size and type described. Chapters 3 and 4 describe the reactors and the reactor coolant systems, Chapter 5 the containment and related systems, and Chapters 6 through 11 the emergency and other auxiliary systems.

Chapter 12 describes I&M's program for organization and training of plant personnel. Chapter 13 contains an outline and description of the initial tests and operations associated with plant startup.

Chapter 14 is a safety evaluation summarizing the analyses which demonstrate the adequacy of the reactor protection system, and the engineered safety features systems. The consequences of various postulated accidents are within the guidelines set forth in the Nuclear Regulatory Commission regulation 10 CFR 100.

by a containment isolation signal derived from the safety injection automatic activation logic and Phase "B" isolation from a containment pressure high-high signal.

f) Reliable on-site diesel-generator power is provided for the engineered safeguards loads in the event of failure of station auxiliary power. In addition, even if external auxiliary power to the station is lost concurrent with an accident, power is available for the engineered safeguards from on-site diesel-generator power to assure protection of the public health and safety for any loss-of-coolant accident.

g) The active components necessary for the proper operation of the engineered safety features are operable from the control room.

The Engineered Safety Features in this plant are the ECCS, the containment structure, the Ice Condenser System, and the Containment Spray System (items a, b, c, d above).

1.3.9 SHARED FACILITIES AND EQUIPMENT

Separate and similar systems and equipment are provided for each unit except as noted below. In those instances where components of a system are shared by both units, those components which are shared are either shown in the following listing or discussed in the applicable Sub-Chapter.

a) Chemical and Volume Control System

Item	Number Shared
Boric Acid Tanks	3
Batching Tank	1
Hold-up Tanks	3
Recirculation Pump	1
Boric Acid Evaporator Feed Pumps	3
Evaporator Feed Ion Exchangers	4
Boric Acid Evaporator (One temporarily in use as a waste evaporator)	2
Monitor Tanks	4
Monitor Tank Pumps	2
Evaporator Condensate Demineralizers	2

b) Spent Fuel Pit Cooling System

Item	Number Shared
Spent Fuel Pool Pumps	2
Spent Fuel Pool Demineralizer	1
Spent Fuel Pool Filter	1
Spent Fuel Pool Heat Exchangers	2
Refueling Water Purification Pump	1

c) Fuel Handling System

Item	Number Shared
Spent Fuel Storage Pool	1
New Fuel Storage Area	1
Decontamination Area	1
Spent Fuel Pool Bridge Crane	1

1.4.4 RELIABILITY AND TESTABILITY OF PROTECTIVE SYSTEMS

Protective systems were designed with a degree of functional reliability and in-service testability which is commensurate with the safety functions to be performed. System design incorporates such features as emergency power availability, preferred failure mode design, redundancy and isolation between control systems and protective systems. In addition, the protective systems were designed such that no single failure would prevent proper system action when required. For design purposes, multiple failures which result from a single event were considered single failures. The proposed criteria of the Institute of Electrical and Electronic Engineers for nuclear power plant protection (IEEE-279) have been utilized in the design of protective systems.

The plant variables monitored and the sensors utilized are identified and discussed at length in Westinghouse proprietary reports submitted in support of the application for an operating license for Donald C. Cook Nuclear Plant and referenced in Chapter 7.

The coincident trip philosophy is carried out to provide a safe and reliable reactor protection system since a single failure will not defeat its function nor cause a spurious reactor trip. Channel independence originates at the process sensor and continues back through the field wiring and containment penetrations to the analog protection racks. The power supplies to the protection sets are fed from instrumentation buses.

Two reactor trip breakers are provided to interrupt power to the control rod drive mechanisms. The breakers main contacts are connected in series. Opening either breaker will interrupt power to all control rod drive mechanisms causing all rods to fall by gravity into the core. Each reactor trip breaker has an undervoltage trip attachment and a shunt trip attachment. Either attachment trips the breaker. Automatic or manual trip initiation activates both the undervoltage and shunt trip attachments. Each protection channel feeds two logic matrices, one for each undervoltage trip circuit.

Each reactor trip channel is designed so that it will go into a trip mode when the channel is de-energized. An open channel or loss of channel power therefore would cause the affected channel to go into a trip mode. Reliability and independence are obtained by redundancy within each channel, except for back-up reactor trips such as the reactor coolant pump breaker position trip. Reactor trip is implemented by interrupting power to the mechanism on each control rod drive mechanism allowing the rod cluster control assemblies (RCCAs) to be inserted by gravity. The protection system is thus inherently safe in the event of a loss of control rod power.

The components of the protective system are designed and laid out so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function will not interfere with that function.

The actuation of the engineered safety features provided for loss-of-coolant accidents (LOCA), e.g., emergency core cooling system and containment spray system, is accomplished from redundant signals derived from reactor coolant system, steam flow, and containment instrumentation. Channel independence originates at the process sensor and is carried through to the analog protection racks. De-energizing a channel will cause that channel to go into its trip mode (See Subchapter 7.5).

A comprehensive program of plant testing is executed for equipment vital to the functioning of engineered safety features. The program consists of performance tests of individual pieces of equipment, and integrated tests of the engineered safety features as a whole, and periodic tests of the actuation circuitry and the performance of mechanical components to assure reliable performance upon demand throughout the plant lifetime.

The following series of periodic tests and checks are conducted to assure that the systems can perform their design functions should they be called on during the plant lifetime.

design provides for periodic testing of active components for operability and required functional performance as well as incorporating provisions to facilitate physical inspection of critical components.

3. Heat removal systems are provided within the containment to cool the containment atmosphere under design basis accident conditions. Two systems of different design principles are provided, the containment spray system and the ice condenser system. These systems have the capacity to adequately cool and reduce the pressure of the containment atmosphere as well as reduce the concentration of halogen fission products.

1.4.8 FUEL AND WASTE STORAGE SYSTEMS

Fuel storage and waste handling facilities are designed such that accidental releases of radioactivity will not exceed the guidelines of 10 CFR 100.

During refueling of the reactor, operations are conducted with the spent fuel under water. This provides visual control of the operation at all times and also maintains low radiation levels. The borated refueling water assures subcriticality and also provides adequate cooling for the spent fuel during transfer. Spent fuel is taken from the reactor core, transferred to the refueling cavity, and placed in the fuel transfer canal. Rod cluster control assembly transfer from a spent fuel assembly to a new fuel assembly is accomplished prior to transferring the spent fuel to the spent fuel storage pool. The spent fuel storage pool is supplied with a cooling system for the removal of the decay heat of the spent fuel. Racks are provided to accommodate the storage of a total of two thousand and fifty fuel assemblies. The storage pool is filled with borated water at a concentration to match that used in the reactor cavity during refueling operations. The spent fuel is stored in a vertical array with sufficient center-to-center

distance between assemblies to assure subcriticality ($k_{eff} \leq 0.95$) even if unborated water were introduced into the pool.[3,4] The water level maintained in the pool provides sufficient shielding to permit normal occupancy of the area by operating personnel. The spent fuel pool is also provided with systems to maintain water cleanliness and to indicate pool water level. Radiation is continuously monitored and a high radiation level is annunciated in the control room.

Water removed from the spent fuel pool must be pumped out as there are no gravity drains. Spillage or leakage of any liquids from waste handling facilities within the auxiliary building go to waste drain system floor drains. These floor drains are connected to separate "contaminated" sumps in the auxiliary building.

Postulated accidents involving the release of radioactivity from the fuel and waste storage and handling facilities are shown in Chapter 14 to result in exposures well within the guidelines of 10 CFR 100.

The refueling cavity, the refueling canal, the fuel transfer canal, and the spent fuel storage pool are reinforced concrete structures with a corrosion resistant liner. These structures have been designed to withstand loads due to postulated earthquakes. The fuel transfer tube, which connects the refueling canal and the fuel transfer canal which forms part of the reactor containment, is provided with a valve and a blind flange which closes off the fuel transfer tube when not in use.

1.4.9 EFFLUENTS

Gaseous, liquid and solid waste disposal facilities have been designed so that the discharge of effluents and off-site shipments are in accordance with applicable governmental regulations.

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AMERICAN ELECTRIC POWER

STATEMENT OF POLICY
FOR THE DONALD C. COOK NUCLEAR PLANT
QUALITY ASSURANCE PROGRAM

POLICY

American Electric Power Company Inc., recognizes the fundamental importance of controlling the design, modification, and operation of Indiana Michigan Power Company's Donald C. Cook Nuclear Plant (Cook Nuclear Plant) by implementing a planned and documented Quality Assurance Program, including Quality Control, that complies with applicable regulations, codes, and standards.

The Quality Assurance Program has been established to control activities affecting safety-related functions of structures, systems, and components in the Cook Nuclear Plant. The Quality Assurance Program supports the goal of maintaining the safety and reliability of the Cook Nuclear Plant at the highest level through a systematic program designed to assure that safety-related items are conducted in compliance with the applicable regulations, codes, standards, and established corporate policies and practices.

As Chairman of the Board and Chief Executive Officer of American Electric Power Company, Inc., I maintain the ultimate responsibility for the Quality Assurance Program associated with the Cook Nuclear Plant. I have delegated functional responsibility for the Quality Assurance Program to the American Electric Power Service Corporation (AEPSC) Senior Executive Vice President-Engineering and Construction. He has, with my approval, delegated further responsibilities as outlined in this statement.

IMPLEMENTATION

The AEPSC Director-Quality Assurance, under the direction of the AEPSC Senior Executive Vice President-Engineering and Construction, has been assigned the overall responsibility for specifying the Quality Assurance program requirements for the Cook Nuclear Plant and verifying their implementation. The AEPSC Senior Executive Vice President-Engineering and Construction has given the AEPSC Director-Quality Assurance authority to stop work on any activity affecting safety-related items that does not meet applicable administrative, technical, and/or regulatory

requirements. The AEPSC Director-Quality Assurance does not have the authority to stop unit operations, but shall notify appropriate plant and/or corporate management of conditions not meeting the aforementioned criteria and recommend that unit operations be terminated.

The AEPSC Vice President-Nuclear Operations, under the direction of the AEPSC Senior Executive Vice President-Engineering and Construction, has been delegated responsibility for effectively implementing the Quality Assurance Program. The AEPSC Vice President-Nuclear Operations is the Manager of Nuclear Operations. All other AEPSC divisions and departments, except Quality Assurance, having a supporting role for the Cook Nuclear Plant are functionally responsible to the Manager of Nuclear Operations.

The Plant Manager, under the direction of the AEPSC Vice President-Nuclear Operations, is delegated the responsibility for establishing the Cook Nuclear Plant Quality Control Program and implementing the Quality Assurance Program at the Cook Nuclear Plant.

The AEPSC Director-Quality Assurance is responsible for providing technical direction to the Plant Manager for matters relating to the Quality Assurance Program at the Cook Nuclear Plant. The AEPSC Director-Quality Assurance is also responsible for maintaining a Quality Assurance Section at the Cook Nuclear Plant to perform required reviews, audits, and surveillances, and to provide technical liaison services to the Plant Manager.

The implementation of the Quality Assurance Program is described in the AEPSC General Procedures (GPs) and subtier department/division procedures, Plant Manager's Instructions (PMIs), and subtier Department Head Instructions and Procedures, which in total document the requirements for implementation of the Program.

Each AEPSC and Cook Nuclear Plant organization involved in activities affecting safety-related functions of structures, systems, and components in the Cook Nuclear Plant has the responsibility to implement the applicable policies and requirements of the Quality Assurance Program. This responsibility includes being familiar with, and complying with, the requirements of the applicable Quality Assurance Program requirements.

COMPLIANCE

The AEPSC Director-Quality Assurance shall monitor compliance with the established Quality Assurance Program. Audit programs shall be established to ensure that AEPSC and Cook Nuclear Plant activities comply with established program requirements, identify deficiencies or noncompliances and obtain effective and timely corrective actions.

Employees engaged in activities affecting safety-related functions of structures, systems, and components in the Cook Nuclear Plant who believe that the Quality Assurance Program is not being complied with, or that a deficiency in quality exists, should notify their supervisor, the AEPSC Director-Quality Assurance, and/or the Plant Manager. If the notification does not in the employee's opinion receive prompt or appropriate attention, the employee should contact successively higher levels of management. Employees reporting such conditions shall not be discriminated against by companies of the American Electric Power System. Discrimination includes discharge or other actions relative to compensation, terms, conditions, or privileges of employment.

R. E. Disbrow
Chairman of the Board and
Chief Executive Officer
American Electric Power Company, Inc.

1.7.1 ORGANIZATION

1.7.1.1 SCOPE

American Electric Power Service Corporation (AEPSC) is responsible for establishing and implementing the Quality Assurance (QA) Program for the operational phase of the Donald C. Cook Nuclear Plant (Cook Nuclear Plant). Although authority for development and execution of various portions of the program may be delegated to others, such as contractors, agents or consultants, AEPSC retains overall responsibility. AEPSC shall evaluate work delegated to such organizations. Evaluations shall be based on the status of safety importance of the activity being performed and shall be initiated early enough to assure effective quality assurance during the performance of the delegated activity.

This section of the Quality Assurance Program Description (QAPD) identifies the AEPSC organizational responsibilities for activities affecting the quality of safety-related nuclear power plant structures, systems, and components, and describes the authority and duties assigned to them. It addresses responsibilities for both attaining quality objectives and for the functions of establishing the QA Program, and verifying that activities affecting the quality of safety-related items are performed effectively in accordance with QA Program requirements.

1.7.1.2 IMPLEMENTATION

1.7.1.2.1 Source of Authority

The Chairman of the Board and Chief Executive Officer of American Electric Power Company, Inc. (AEP) and AEPSC is responsible for safe operation of the Cook Nuclear Plant. Authority and responsibility for effectively implementing the QA Program for plant modifications, operations and maintenance are delegated through the AEPSC Senior Executive Vice President - Engineering and Construction, to the AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations).

In the operation of a nuclear power plant, the licensee is required to establish clear and direct lines of responsibility, authority and accountability. This requirement is applicable to the organization providing support to the plant, as well as to the plant staff.

The AEPSC corporate support of the Cook Nuclear Plant is the responsibility of the entire organization under the direction of the Manager of Nuclear Operations who maintains primary responsibility for the Cook Nuclear Plant within the corporate organization. The AEPSC Vice President - Nuclear Operations is the Manager of Nuclear Operations. All other AEPSC divisions and departments, other than the Quality Assurance Division, having a supporting role for nuclear operations and for the Cook Nuclear Plant are functionally responsible to the Manager of Nuclear Operations (reference Figure 1.7-1).

In order to facilitate a more thorough understanding of the support functions, some of the responsibilities, authorities, and accountabilities within the organization are as follows:

1) The responsibilities of the Manager of Nuclear Operations shall be dedicated to the area of Cook Nuclear Plant operations and support.

2) The Manager of Nuclear Operations shall be responsible for, and has the authority to direct, all Cook Nuclear Plant operational and support matters within the corporation and shall make, or concur, in all final decisions regarding significant nuclear safety matters.

3) AEPSC organization managers responsible for Cook Nuclear Plant matters shall be familiar with activities within their scope of responsibility that affect plant safety and reliability. They shall be cognizant of, and sensitive to, internal and external factors that might affect the operations of Cook Nuclear Plant.

4) AEPSC organization managers responsible for Cook Nuclear Plant matters have a commitment to seek and identify problem areas and take corrective action to eliminate unsafe conditions, or to improve trends that will upgrade plant safety and reliability.

5) The Manager of Nuclear Operations shall ensure that Cook Nuclear Plant personnel are not requested to perform inappropriate work or tasks by corporate personnel, and shall control assignments and requests that have the potential for diverting the attention of the Plant Manager from the primary responsibility for safe and reliable plant operation.

6) AEPSC organization managers having Cook Nuclear Plant support responsibilities, as well as the Plant Manager and plant organization managers, shall be familiar with the policy statements from higher management concerning nuclear safety and operational priorities. They shall be responsible for ensuring that activities under their direction are performed in accordance with these policies.

1.7.1.2.2 <u>Responsibility for Attaining Quality Objectives in AEPSC Nuclear Operations</u>

The AEP Chairman of the Board and Chief Executive Officer has delegated the functional responsibility of the Quality Assurance Program to the AEPSC Senior Executive Vice President - Engineering and Construction.

The AEPSC Director - Quality Assurance, under the direction of the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for specifying QA Program requirements and verifying their implementation.

The AEPSC Vice President - Nuclear Operations, under the direction of the AEPSC Senior Executive Vice President - Engineering and

Construction, is responsible for effectively implementing the QA Program.

The Plant Manager, under the direction of the AEPSC Vice President - Nuclear Operations, is responsible for establishing the Cook Nuclear Plant Quality Control Program and implementing the QA Program at the Cook Nuclear Plant.

[illegible]

AEPSC has an independent off-site Nuclear Safety and Design Review Committee (NSDRC) which has been established pursuant to the requirements of the Technical Specifications for the Cook Nuclear Plant. The function of the NSDRC is to oversee the engineering, design, operation, and maintenance of the Cook Nuclear Plant by performing audits and independent reviews of activities which are specified in the facility Technical Specifications.

The Cook Nuclear Plant on-site review group is the Indiana Michigan Power Company (I&M) Plant Nuclear Safety Review Committee (PNSRC). This committee has also been established pursuant to the requirements of the Cook Nuclear Plant Technical Specifications. The function of the PNSRC is to review plant operations on a continuing basis and advise the Plant Manager on matters related to nuclear safety.

1.7.1.2.3 Corporate Organization

American Electric Power Company

AEP, the parent holding company, wholly owns the common stock of all AEP System subsidiary (operating) companies. The major operating companies and generation subsidiaries are shown in Figure 1.7-2. The Chairman of the Board and Chief Executive Officer of AEP is the Chief Executive Officer of AEPSC and all operating companies. The responsibility for the functional management of the major operating companies is vested in

the President of each operating company reporting to the AEPSC President and Chief Operating Officer who reports to the AEPSC Chairman of the Board.

American Electric Power Service Corporation

The responsibility for administrative and technical direction of the AEP System and its facilities is delegated to AEPSC. AEPSC provides management and technological services to the various AEP System companies.

Operating Companies

The operating facilities of the AEP System are owned and operated by the respective operating companies. The responsibility for executing the engineering, design, construction, specialized technical training, and certain operations' supervision is vested in AEPSC, while all, or part, of the administrative functional responsibility is assigned to the operating companies. In the case of Cook Nuclear Plant, I&M general office staff (headquarters) provides public affairs, accounting, industrial safety direction and procurement support.

The Cook Nuclear Plant is owned and operated by I&M which is part of the AEP System.

1.7.1.2.4 Quality Assurance Responsibility of AEPSC

1) AEPSC provides the technical direction for the Cook Nuclear Plant, and as such makes the final decisions pertinent to safety-related changes in plant design. Further, AEPSC reviews Nuclear Regulatory Commission (NRC) letters, bulletins, notices, etc., for impact on plant design, and the need for design changes or modifications.

2) AEPSC furnishes quality assurance, engineering, design, construction, licensing, NRC correspondence, fuel management and radiological support activities.

3) AEPSC provides additional service in matters such as supplier qualification, procurement of original equipment and replacement parts, and the process of dedicating commercial grade items or services to safety-related applications.

4) The AEPSC QA Division provides technical direction in quality assurance matters to AEPSC and the Cook Nuclear Plant, and oversees the adequacy, effectiveness and implementation of the QA Program through review and audit activities.

5) Cognizant Engineer - (e.g., System Engineer, Equipment Engineer, Lead Engineer, Responsible Engineer, etc.) - The cognizant engineer, and/or engineer with the other titles noted, is that AEPSC individual who provides the engineering/design expertise for a particular area of responsibility. This responsibility includes the implementation of the quality assurance and quality control measures for systems, equipment, structures, or functional areas included in that individual's responsibility. The various titles used for the identification of an individual's responsibility and assignment shall be understood to mean the same as cognizant engineer in the respective areas of responsibility.

<u>Quality Assurance Responsibility of I&M - Cook Nuclear Plant</u>

I&M's Cook Nuclear Plant staff operates the Cook Nuclear Plant in accordance with licensing requirements, including the Technical Specifications and such other commitments as established by the operating licenses. The Plant Manager Instruction (PMI) system and subtier instructions and procedures describe the means by which compliance is achieved and responsibilities are assigned, including interfaces with AEPSC. Figure 1.7-3 indicates the organizational

relationships within the AEP System pertaining to the operation and support of the Cook Nuclear Plant.

1.7.1.2.5 Organization (AEPSC)

The Chairman of the Board and Chief Executive Officer is ultimately responsible for the QA Program associated with the Cook Nuclear Plant. This responsibility has been functionally delegated to the AEPSC Senior Executive Vice President - Engineering and Construction. The AEPSC Senior Executive Vice President - Engineering and Construction has further delegated responsibilities which are administered through the following AEPSC organization management personnel:

- AEPSC Director - Quality Assurance
- AEPSC Vice President - Nuclear Operations
- AEPSC Vice President - Project Management and Construction

Quality Assurance Division

The AEPSC Director - Quality Assurance, reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for the Quality Assurance Division (QAD). The QAD consists of the following sections (Figure 1.7-4):

- Quality Assurance Engineering Section
- Nuclear Software Quality Assurance Section
- Audits and Procurement Section
- Quality Assurance Support Section
- Quality Assurance Section (Site)

The QAD is organizationally independent and is responsible to perform the following:

- Specify QA Program requirements.
- Identify quality problems.
- Initiate, recommend, or provide solutions through designated channels.

- Verify implementation of solutions, as appropriate.
- Prepare, issue and maintain QA Program documents, as required.
- Verify the implementation of the QA Program through scheduled audits and surveillances.
- Verify the implementation of computer software quality assurance through reviews, surveillances and audits.
- Audit engineering, design, procurement, construction and operational documents for incorporation of, and compliance with, applicable quality assurance requirements to the extent specified by the AEPSC management-approved QA Program.
- Organize and conduct the QA auditor orientation, training, certification and qualification of AEPSC audit personnel.
- Provide direction for the collection, storage, maintenance, and retention of quality assurance records.
- Maintain, on data base, a list of suppliers of nuclear (N) items and services, plus other selected categories of suppliers.
- Identify noncompliances of the established QA Program to the responsible organizations for corrective actions, and report significant occurrences that jeopardize quality to senior AEPSC management.
- Follow up on corrective actions identified by QA during and after disposition implementation.
- Review the disposition of conditions adverse to quality to assure that action taken will preclude recurrence.
- Conduct in-process QA audits or surveillances at supplier's facilities, as required.
- Assist and advise other AEP/AEPSC groups in matters related to the QA Program.
- Conduct audits as directed by the NSDRC.
- Review AEPSC investigated Problem Reports and associated corrective and preventive action recommendations.
- Maintain cognizance of industry and governmental quality assurance requirements such that the QA Program is compatible with requirements, as necessary.

- Recommend for revision to, or improvements in, the established QA Program to senior AEPSC management.
- Audit dedication plans for commercial grade items and services.
- Issue "Stop Work" orders when significant conditions adverse to safety-related items are identified to prevent unsafe conditions from occurring and/or continuing.
- Provide AEPSC management with periodic reports concerning the status, adequacy and implementation of the QA Program.
- Prepare and conduct special verification and/or surveillance programs on in-house activities, as required or requested.
- Routinely attend, and participate in, daily plant work schedule and status meetings.
- Provide adequate QA coverage relative to procedural and inspection controls, acceptance criteria, and QA staffing and qualification of personnel to carry out QA assignments.
- Determine the acceptability of vendors to supply products and services for safety related applications.

Amplification of Specific Responsibilities

- Qualification of the AEPSC Director - Quality Assurance

  The AEPSC Director - Quality Assurance shall possess the following position requirements:

  - Bachelor's degree in engineering, scientific, or related discipline.
  - Ten (10) years experience in one of, or a combination of, the following areas: engineering, design, construction, operations, maintenance of fossil or nuclear power generation facilities' or utility facilities' QA, of which at least four (4) years must be experience in nuclear quality assurance related activities.
  - Knowledge of QA regulations, policies, practices and standards.
  - The same, or higher, organization reporting level as the highest line manager directly responsible for performing activities affecting the quality of

safety-related items, such as engineering, procurement, construction and operation, and is sufficiently independent from cost and schedule.

- Effective communication channels with other senior management positions.
- Responsibility for approval of QA Manual(s).
- Performance of no other duties or responsibilities unrelated to QA that would prevent full attention to QA matters.

- <u>Stop Work Orders</u>

The AEPSC QAD is responsible for ensuring that activities affecting the quality of safety-related items are performed in a manner that meets applicable administrative, technical, and regulatory requirements. In order to carry out this responsibility, the AEPSC Senior Executive Vice President - Engineering and Construction has given the AEPSC Director - Quality Assurance the authority to stop work on any activity affecting the quality of safety-related items that does not meet the aforementioned requirements. Stop work authority has been further delegated by the AEPSC Director - Quality Assurance to the AEPSC Quality Assurance Superintendent (site).

The AEPSC Director - Quality Assurance and the AEPSC Quality Assurance Superintendent do not have the authority to stop unit operations, but will notify appropriate Cook Nuclear Plant and/or corporate management of conditions which do not meet the aforementioned criteria, and recommend that unit operations be terminated.

- QA Auditor, Qualification and Certification Program

- AEPSC has established and maintains a QA auditor training and certification program for all AEPSC QA auditors.

- Problem Identification, Reporting and Escalation

- AEPSC has established mechanisms for the identification, reporting and escalation of problems affecting the quality of safety-related items to a level of management whereby satisfactory resolutions can be obtained.

Nuclear Operations

The AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations), reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for nuclear operations. Nuclear operations is comprised of the AEPSC Nuclear Operations Department, the AEPSC Nuclear Engineering Department, and the Cook Nuclear Plant Organization.

The AEPSC Nuclear Operations Department (NOD) is responsible for the following:

- Formulate policies and practices relative to safety, licensing, operation, maintenance, fuel management, and radiological support.
- Provide the Plant Manager with the technical and managerial guidance, direction and support to ensure the safe operation of the plant.
- Maintain liaison with the AEPSC Director - Quality Assurance.
- Implement the requirements of the AEPSC QA Program.
- Maintain knowledge of the latest safety, licensing, and regulatory requirements, codes, standards, and federal regulations applicable to the operation of Cook Nuclear Plant.

- Accomplish the procurement, economic, technical, licensing and quality assurance activities dealing with the reactor core and its related fuel assemblies and components.
- Prepare bid specifications, evaluate bids, and negotiate and administer contracts for the procurement of all nuclear fuel and related components and services.
- Maintain a special nuclear material accountability system.
- Provide analyses to support nuclear steam supply system operation,including reactor physics, fuel economics, fuel mechanical behavior, core thermal hydraulic and LOCA and non-LOCA transient safety analysis and other analysis activities as requested, furnish plant Technical Specification changes and other licensing work, and participate in NRC and NSDRC meetings as required by these analyses.
- Perform reactor core operation follow-up activities and other reactor core technical support activities as requested, and arrange for support from the fuel fabricator, when needed.
- Contract for, and provide technical support for, disposal of both high level and low level radioactive waste.
- Coordinate the development of neutronics and thermal hydraulic safety codes and conduct safety analyses.
- Conduct studies of the Cook Nuclear Plant licensing bases to determine the optimal changes to support unit operations at a lower primary pressure and temperature.
- Coordinate NOD computer code development, and provide the interface control for NOD with the AEPSC Information System Department and Cook Nuclear Plant.
- Obtain and maintain the NRC Operating License and Technical Specifications for the Cook Nuclear Plant.
- Act as the communication link between the NRC, AEPSC, and the plant staff.
- Perform and coordinate efforts involved in gathering information, performing calculations and generic studies; preparing criteria, reports, and responses; reviewing items affecting safety; and interpreting regulations.

- Review, coordinate, and resolve all matters pertaining to nuclear safety between Cook Nuclear Plant and AEPSC. This includes, but is not limited to: the review of certain plant design changes to ensure that the requirements of 10CFR50.59 are met; the preparation of safety evaluations, or reviews, for any designated subject; the preparation of changes to, and appropriate interpretation of, the plant Technical Specification submittals of license amendments; and the analysis of plant compliance with regulatory requirements.
- Primary corporate contact for most oral and written communication with the NRC.
- Provide support in key areas of expertise, such as nuclear engineering, probabilistic analysis, thermohydraulic analysis, chemical engineering, mechanical engineering, electrical engineering, and technical writing.
- Interface with vendors and other outside organizations on matters connected with the nuclear steam supply system and other areas affecting the safe design and operation of nuclear plants.
- Participate, as appropriate, in the review of nuclear plant operating experiences, and relate those experiences to the design and safe operation of Cook Nuclear Plant.
- Review, evaluate, and respond to NRC requests for information and NRC notifications of regulatory changes resulting in plant modifications or new facilities. Such responses are generated in accordance with appropriate AEPSC Administrative Procedures.
- Develop, specify, and/or review conceptual nuclear safety criteria for Cook Nuclear Plant in accordance with established regulations. This includes all information contained in the FSAR, as well as specialized information such as environmental qualification and seismic criteria.
- Review and evaluate performance requirements for systems, equipment and materials for compliance with specified safety criteria.
- Review, on a conceptual basis, plant reports and proposed plant safety-related design changes, to the extent that they are related

improvements, the ALARA program, the radiation monitoring system, the environmental radiological monitoring and sampling program, dose and shielding analysis, radiochemistry review, implications of federal regulations, and meteorological monitoring.

- Cook Nuclear Plant and corporate emergency planning, including procedure development, exercise scheduling, facility procurement and maintenance, and the liaison with off-site emergency planning groups, such as FEMA and the Michigan State Police.
- Review federal codes and regulations as they relate to the development, implementation, revision and distribution of the Modified Amended Security Plan (MASP).
- Interface with the plant's security department providing support for the security plan, reviewing security facilities, maintaining security document files, and developing the employee fitness for duty/background screening program.
- Provide Nuclear General Employee Training (NGET) for AEPSC personnel.
- Coordinate the development of training for AEPSC personnel who support the operation and maintenance of Cook Nuclear Plant, ensuring a unified training program meeting annual goals and objectives.
- Participate on the ALARA committee.
- Prepare responses to the NRC on radiological, emergency planning and security issues.
- Serve as technical advisors on plant audits.
- Remain cognizant of current decommissioning practices and developments.

AEPSC Nuclear Engineering Department

The AEPSC Chief Nuclear Engineer, reporting to the AEPSC Vice President - Nuclear Operations, is responsible for certain engineering and design functions. The AEPSC Nuclear Engineering Department is comprised of engineering and design entities at AEPSC, as well as an

to the ultimate safe operation of the plant, for compliance with safety regulations, plant Technical Specifications, the Updated FSAR design basis, and with any other requirements under the Operating License, to determine if there are any unreviewed safety questions as defined in 10CFR50.59.

- Perform reviews of Problem Reports and 10CFR21 reviews in accordance with corporate requirements.
- Operate the Action Item Tracking System (AIT) for AEPSC internal commitment tracking.
- Coordinate design changes for the Cook Nuclear Plant, acting as a focal point within AEPSC. This program primarily involves project management responsibilities for scheduling and implementing Request for Changes (RFCs), and includes extensive interfacing with engineering, design, construction, and Cook Nuclear Plant.
- Provide working-level coordination with the Institute of Nuclear Power Operations (INPO) in the areas of INPO training, seminars, and workshops. This effort includes providing AEPSC access to INPO resources, such as NUCLEAR NETWORK and Nuclear Plant Reliability Data System (NPRDS), and effectively integrating AEPSC and Cook Nuclear Plant efforts towards utilizing INPO recommendations contained in operating experience reports to improve Cook Nuclear Plant performance.
- Coordinate daily communication with the Cook Nuclear Plant, provide AEPSC management with a daily plant status report, and make presentations to senior management at regularly scheduled construction staff meetings.
- Process incoming vendor information.
- Coordinate operations within AEPSC that support the Cook Nuclear Plant Facility Data Base (FDB).
- Contribute to the annual FSAR updates through reviews of Licensee Event Reports, design changes and the Annual Operating Report.
- Radiological, emergency and security planning.
- Corporate support of the Cook Nuclear Plant's radiation protection and health physics program, technical reviews and advice on the radiological aspects of design changes, modifications or capital

on-site engineering and design support organization at the Cook Nuclear Plant.

The AEPSC Civil Engineering, Electrical Engineering, and System Planning Departments provide periodic, technical assistance for the Cook Nuclear Plant. The administrative and quality assurance controls for this assistance are controlled through documented interface agreements with the AEPSC Nuclear Engineering Department.

AEPSC Nuclear Engineering Department (NED) is responsible for the following:

- Provide planning, engineering and design of the electrical facilities inside Cook Nuclear Plant up to the high voltage (HV) bushings of the main generator transformers and mechanical facilities inside Cook Nuclear Plant including:

    * determination of general layout and design;
    * selection of equipment;
    * preparation of one-line and flow diagrams; and,
    * coordination of inside and outside plant facilities.

- Provide engineering and design of all controls for operation and protection of nuclear steam supply, steam generator, turbine generator, auxiliary equipment and general plant protection, including checking and approving elementary, one-line, and flow drawings.
- Ensure that all purchased equipment conforms to accepted standards and fulfills the desired function.
- Closely follow manufacturer's engineering and design processes to assure provision of adequate and reliable equipment upon which depend the safety, reliability, and performance of the unit and plant.
- Prepare, review and/or approve design changes, sketches, drawings, calculations, and design verifications, as required.

- Prepare and/or approve dedication plans, specifications and purchase requisitions.
- Perform drawing review of equipment, as appropriate.
- Develop, review and/or approve procedures or correspondence as appropriate.
- Obtain, review and perform engineering and design evaluations, including environmental equipment qualification (EQ).
- Establish and maintain a central file for equipment environmental qualification documentation.
- Perform calculations for proper application of equipment.
- Perform and evaluate investigations, analyses and reports for facilities pertaining to the engineering design, operation and maintenance of the Cook Nuclear Plant.
- Assist field personnel in installation, start-up, and subsequent locating of problems in equipment, and in determining proper operation of equipment, during normal or after emergency operations.
- Maintain a constant awareness for improvements and more reliable design of equipment and facilities, maintenance and operating methods or procedures.
- Maintain a constant awareness of activities to ensure compliance with all applicable policies and procedures, initiating, when required, training or retraining programs.
- Participate, as assigned, on the NSDRC and NSDRC subcommittees, and participate in matters covered in the committee's charter.
- Provide responses to NRC correspondence, as required.
- Participate in the evaluation and remedy of any situation requiring activation of the emergency response organization.
- Provide support personnel for the Emergency Response Organization.
- Provide technical support in areas of operation and maintenance, including: the Inservice Inspection (ISI) Program; the QA Program; the Fire Protection QA Program; the AEP ALARA Program covering radiation protection; and, the corporate and plant Industrial Safety program.

- Provide technical direction and assistance in the layout and arrangement of equipment piping, systems, controls, etc., for the development of drawings.
- Develop System Descriptions.
- Provide analytical support in engineering and design disciplines (e.g., heat transfer, thermodynamics, fluid dynamics).
- Provide engineering and design evaluations for PRs, LERs, INPO SOERs, and NRC Bulletins.
- Participate, as assigned, on the AEPSC Problem Assessment Group (PAG).
- Make recommendations and assist in the formulation of policies and practices relating to the design and engineering of office and service buildings, miscellaneous structures and material handling equipment, and provide the general supervision of the engineering of such facilities, structures and equipment.
- Initiate and/or review, approve and control laboratory and field investigations and feasibility studies.
- Prepare and administer equipment, labor and service contracts.
- Arrange for outside engineering, design and consulting assistance, as required.
- Perform shop and field surveillance on equipment being manufactured, fabricated, or installed.
- Provide field services to the Cook Nuclear Plant, including the assigning of personnel, as are required, during construction, normal or forced outages, or as requested.
- Assist in the planning and execution of maintenance work on equipment, facilities, buildings and other structures.
- Supervise maintenance and repairs of all masonry and concrete work at Cook Nuclear Plant, including supplying qualified inspection personnel.
- Direct testing of materials used in concrete and testing of soils to be used in work at the Cook Nuclear Plant.
- Review and recommend concrete mix formulations for all new construction.

- Implement the corrective action program, with regard to activities affecting the quality of safety-related items and services, that controls and documents items, services or activities which do not conform to requirements.
- Assist in the preparation of applications for federal, state and local permits relative to installations being made which require such permits.
- Conduct periodic management reviews of the activities of the department to ensure compliance with the objectives of the QA Program, and external technical surveillance, as necessary, of consultants, outside organizations and vendors over which the department is cognizant.
- Establish and maintain a file for QA records.
- Develop, review and approve designs and drawings for mechanical, electrical and structural systems, equipment and facilities of the Cook Nuclear Plant.
- Perform required calculations and analyses, including pipe stress, pipe support design, cable sizing, conduit and cable tray support and structural steel and concrete.
- Assist field personnel in the resolution of problems stemming from the installation of design changes, or from as-found plant conditions, including assigning personnel to the plant.
- Formulate, administer, and implement policies and practices relating to the engineering, and design of the Cook Nuclear Plant.
- Conduct functions so as to be in conformance with the operating licenses of the Cook Nuclear Plant.
- Investigate evaluate and correct problems.
- Coordinate special projects and studies, as required.
- Coordinate the development and maintenance of the computerized Design Drawing Control (DDC) and the Vendor Drawing Control (VDC) programs which include coordinating the programs with interfacing divisions/departments.
- Control the issuance and distribution of drawings for the Cook Nuclear Plant, including monitoring of the Aperture Card Microfilm Program.

- Supervise and control the work of consultants, architect/engineers and outside engineering and design agencies supplying services to AEPSC in their discipline and process notification of defects in accordance with company requirements. Also perform detailed reviews of engineering and design work submitted by outside agencies.
- Review and update applicable sections of Cook Nuclear Plant Updated FSAR as assigned.
- Participate, as members and as assigned, on committees and ad hoc task forces that review nuclear activities.

Cook Nuclear Plant

The Plant Manager reports functionally and administratively to the AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations) and is responsible for the Cook Nuclear Plant activities (Figure 1.7-5).

The Cook Nuclear Plant organization is responsible for the following:

- Ensure the safety of all facility employees and the general public relative to general plant safety, as well as radiological safety, by maintaining strict compliance with plant Technical Specifications, procedures and instructions.
- Recommend facility engineering modification and initiate and approve plant improvement requisitions.
- Ensure that work practices in all plant departments are consistent with regulatory standards, safety, approved procedures, and plant Technical Specifications.
- Provide membership, as required, on the PNSRC.
- Maintain close working relationships with the NRC, as well as local, state, and federal government regulatory officials regarding conditions which could affect, or are affected, by Cook Nuclear Plant activities.
- Set up plant load schedules and arrange for equipment outages.

- Develop and efficiently implement all site centralized training activities.
- Administer the centralized facility training complex, simulator, and programs ensuring that program development is consistent with the systematic approach to training, maintain INPO accreditations, regulatory and corporate requirements.
- Ensure that human resource activities include employee support programs (i.e., fitness for duty) consistent with INPO/NUMARC guidelines, company policies, and regulatory requirements and standards.
- Administer the NRC approved physical Security Program in compliance with regulatory standards, Modified Amended Security Plan (MASP), and company policy.
- Supervise, plan, and direct the activities related to the maintenance and installation of all Cook Nuclear Plant equipment, structures, grounds, and yards.
- Prepare and maintain records and reports pertinent to equipment maintenance and regulatory agency requirements.
- Administer contracts and schedule outside contractors' work forces.
- Enforce and coordinate Cook Nuclear Plant regulations, procedures, policies, and objectives to assure safety, efficiency, and continuity in the operation of the Cook Nuclear Plant within the limits of the operating license and the Technical Specifications and formulation of related policies and procedures.
- Plan, schedule, and direct activities relating to the operation of the Cook Nuclear Plant and associated switchyards; cooperate in planning and scheduling of work and procedures for refueling and maintenance of the Cook Nuclear Plant; and direct and coordinate fuel loading operations.
- Review reports and records, direct general inspection of operating conditions of plant equipment, and investigate any abnormal conditions, making recommendations for repairs. Establish and administer equipment clearance procedures consistent with company, plant, and radiation protection standards; authorize and arrange

for equipment outages to meet normal or emergency conditions. Provide the shift operating crews with appropriate procedures and instructions to assist them in operating the Cook Nuclear Plant safely and efficiently.

- Approve operator training programs administered by the Cook Nuclear Plant Training Department designed to provide operating personnel with the knowledge and skill required for safe operation of the facility, and for obtaining and holding NRC operator licenses. Coordinate training programs in plant safety and emergency procedures for Cook Nuclear Plant Operating Department personnel to ensure that each shift group will function properly in the event of injury of personnel, fire, nuclear incident, or civil disorder.
- Advance planning and overall conduct of scheduled and forced outages, including the scheduling and coordination of all plant activities associated with refueling, preventive maintenance, corrective maintenance, equipment overhaul, Technical Specification surveillance, and design change installations.
- Coordinate all Cook Nuclear Plant activities associated with the initiation, review, approval, engineering, design, production, examination, inspection, test, turnover, and close out of design changes.
- Develop and implement an effective Quality Control (QC) Program. This encompasses, but is not limited to, the planning and directing of quality control activities to assure that industry codes, NRC regulations, and company instructions and policies regarding quality control for Cook Nuclear Plant are implemented, qualified personnel perform the work, and that these activities are properly documented.
- Prepare reports of reportable events which are mandated by the NRC and the Technical Specifications.
- Direct the activities of contractor QC/nondestructive examination (NDE) personnel assigned to the Safety and Assessment Department and provide inspections of work performed.

- Prepare statistical reports utilized in NRC Appraisal Meetings and Enforcement Conference.
- Coordinate the efforts of outside agencies, such as American Nuclear Insurers (ANI), INPO, and third-party inspector programs.
- Maintain knowledge of developments and changes in NRC requirements, industry standards and codes, regulatory compliance activities, and quality control disciplines and techniques.
- Stop plant operation in the event that conditions are found which are in violation of the Technical Specifications or adverse to quality.
- Maintain and renew accreditation of training programs.
- Qualification and certification of I&M personnel performing inspections or tests of major modifications and non-routine maintenance to the requirements of Regulatory Guide 1.5 and ANSI N45.2.6, except as noted in Appendix B hereto, item 9.
- Qualification and certification of I&M NDE personnel to the requirements of the AEP NDE Manual.
- Qualification of I&M personnel performing inspection of normal operating activities to ANSI N18.1.
- Proper certification of contractor inspection, test and examination personnel in accordance with Regulatory Guide 1.5, ANSI N45.2.6, ASME B&PV Code and/or SNT-TC-1A, as applicable.
- Perform peer inspections of work completed by I&M personnel by independent persons qualified to ANSI N18.7.
- Conduct of the Inservice Inspection (ISI) Program.
- Procurement, receiving, quality control receipt inspection, storage, handling, issue, stock level maintenance, and overall control of stores items.
- Provide material service and support in accordance with policies and procedures required by AEP Purchasing and Stores, AEPSC QA, and the NRC, which are administered and enforced in a total effort to ensure safety and plant reliability.
- Plan and direct engineering and technical studies, nuclear fuel management, equipment performance, instrument and control maintenance, on-site computer systems, Shift Technical Advisors,

and emergency planning for the Cook Nuclear Plant. These activities support daily on-site operations in a safe, reliable, and efficient manner in accordance with all corporate policies, applicable laws, regulations, licenses, and Technical Specification requirements.

- Implement station performance testing and monitor programs to ensure optimum plant efficiency.
- Direct programs related to on-site fuel management and reactor core physics testing, and ensure satisfactory completion.
- Establish testing and preventive maintenance programs related to station instrumentation, electrical systems, and computers.
- Recommend alternatives to Cook Nuclear Plant operation, technical or emergency procedures, and design of equipment to improve safety of operations and overall plant efficiency.
- Implement the corporate Emergency Plan as it pertains to the Cook Nuclear Plant site.
- Provide technical and engineering services in the fields of chemistry, radiation protection, ALARA, and environmental in support of the safe operation of the plant and the health and safety of the employees and the public.
- Plan and schedule the activities of the Radiation Protection Department of the Cook Nuclear Plant in support of operations and maintenance.
- Establish chemistry, radiochemistry, and health physics criteria which ensure maximum equipment life, and the protection of the health and safety of the workers and the public.
- Establish sampling and analysis programs which ensure the chemistry, radiochemistry, and health physics criteria are within the established criteria.
- Establish and direct investigations, responses, and corrective actions when outside the established criteria.
- Administer and direct the Cook Nuclear Plant's radioactive waste programs, including volume reduction, packaging and shipping.
- Administration of the QA Records Program.
- Maintain the Cook Nuclear Plant Facility Data Base.

Project Management and Construction Department

The AEPSC Vice President - Project Management and Construction, reporting to the AEPSC Senior Executive Vice President - Engineering and Construction, is responsible for the Project Management and Construction Department.

Reporting to the AEPSC Vice President - Project Management and Construction are the following:

- Site Construction Manager, reporting administratively to the AEPSC Vice President - Project Management and Construction, and functionally to the Plant Manager - Cook Nuclear Plant.

The Project Management and Construction Department is responsible for the following:

- Administer and implement construction job orders issued by the Cook Nuclear Plant organization for major modifications,replacement and maintenance work with outside contractors.
- Administer and monitor contractor's industrial safety programs and performance.
- Administer human resources' functions for site construction organization.
- Manage construction labor relations with the International Building and Construction Trades Unions.
- Scope, bid, recommend awards and administer construction labor and services contracts.
- Plan, organize and control major construction projects, as assigned by the AEPSC Senior Executive Vice President - Engineering and Construction.
- Maintain cognizance on matters pertaining to the Cook Nuclear Plant and corporate emergency response organization.
- Prepare construction labor estimates.

- Provide constructability guidance when requested in support of engineering and design changes.
- Participate on the Nuclear Safety Design Review Committee.

Purchasing and Stores Department (not charted)

The AEPSC Executive Vice President - Operations, reporting to the AEPSC President and Chief Operating Officer, is responsible for the Purchasing and Stores Department through the AEPSC Vice President - Purchasing and Materials Management.

The Purchasing and Stores Department is responsible for the following:

- Procurement of safety-related items from only qualified and approved suppliers.
- Provide supervision to Cook Nuclear Plant Purchasing Section.
- Provide ordering and stocking descriptions of safety-related items and include these descriptions in the Cook Nuclear Plant inventory catalog, including necessary communications with suppliers, cognizant engineers, the Cook Nuclear Plant Stores Supervisor and other appropriate personnel.
- Maintain and control the Material & Equipment database file for safety-related items.
- Establish computerized inventory status reports, on line inventory and purchase order inquiry capabilities and other procedures to order, track and control materials.
- Coordinate procurement activities with AEPSC Nuclear Operations, AEPSC Nuclear Engineering, Cook Nuclear Plant Site Purchasing, AEPSC Quality Assurance and Cook Nuclear Plant personnel.
- Prepare and issue requests for quotations, contracts, service orders and purchase orders for safety-related items.
- Establish a system to implement corrective action as described in the AEPSC General Procedures for the Cook Nuclear Plant.
- Establish a system of document keeping and transmittal.
- Establish a system of document control for controlled procedures, instructions, and purchasing documents for safety-related items.

- The maintenance and control of selected standard procurement document phrases as identified by the Director - Quality Assurance, or designee.
- Conduct training sessions involving purchasing personnel and others on an annual basis, or more frequently, as required, and ascertain that training sessions include complete responsibilities associated with the purchase of safety-related items.

1.7.2 QUALITY ASSURANCE PROGRAM

1.7.2.1 SCOPE

Policies that define and establish the Cook Nuclear Plant QA Program are summarized in the individual sections of this document. The program is implemented through procedures and instructions responsive to provisions of the QAPD, and will be carried out for the life of the Cook Nuclear Plant.

Quality assurance controls apply to activities affecting the quality of safety-related structures, systems and components to an extent based on the importance of those structures, systems, components, etc., (items) to safety. Such activities are performed under controlled conditions, including the use of appropriate equipment, environmental conditions, assignment of qualified personnel, and assurance that all applicable prerequisites have been met.

Safety-related items are defined as items:

- Which are associated with the safe shutdown (hot) of the reactor; or isolation of the reactor; or maintenance of the integrity of the reactor coolant system pressure boundary.

OR

- Whose failure might cause or increase the severity of a design basis accident as described in the Updated FSAR; or lead to a release of radioactivity in excess of 10CFR100 guidelines.

In general, items are classified as safety-related if they are: Seismic Class I, or Electrical Class 1E; or associated with the Engineered Safety Features Actuation System (ESFAS); or associated with the Reactor Protection System (RPS).

A special QA Program has been implemented for Fire Protection items (Section 1.7.19 herein).

The QA Program also includes provision for Radwaste QA in accordance with the requirements of 10CFR71, part H.

QA Program status, scope, adequacy, and compliance with 10CFR50, Appendix B, are regularly reviewed by AEPSC management through reports, meetings, and review of audit results.

The implementation of the QA Program may be accomplished by AEPSC and/or Indiana Michigan Power Company or delegated in whole or in part to other AEP System companies or outside parties. However, AEPSC and/or Indiana Michigan Power Company retain full responsibility for all activities affecting safety-related items. The performance of the delegated organization is evaluated by audit or surveillances on a frequency commensurate with their scope and importance of assigned work.

1.7.2.2 **IMPLEMENTATION**

1.7.2.2.1

The Chairman of the Board and Chief Executive Officer of AEPSC has stated in a signed, formal "Statement of Policy", that it is the corporate policy to comply with the provisions of applicable codes, standards and regulations pertaining to quality assurance for nuclear power plants as required by the Cook Nuclear Plant operating licenses.

The statement makes this QAPD and the associated implementing procedures and instructions mandatory, and requires compliance by all responsible organizations and individuals. The statement also identifies the

management positions within the companies vested with responsibility and authority for implementing the program and assuring its effectiveness.

1.7.2.2.2

The QA Program at AEPSC and the Cook Nuclear Plant consist of controls exercised by organizations responsible for attaining quality objectives, and by organizations responsible for assurance functions.

The QA Program effectiveness is continually assessed through management review of various reports, NSDRC review of the QA audit program, and shall also be periodically reviewed by independent outside parties as deemed necessary by management.

The QA Program described in this QAPD is intended to apply for the life of the Cook Nuclear Plant.

The QA Program applies to activities affecting the quality of safety-related structures, components, and related consumables during plant operation, maintenance, testing, and all design changes. Safety-related structures, systems and components are identified in the Facility Data Base and other documents which are developed and maintained for the plant.

As deemed necessary by the AEPSC Director - Quality Assurance, or the Plant Manager, applicable portions of the QA Program controls will be applied to nonsafety-related activities associated with the implementation of the QA Program to ensure that commitments are met (e.g., off-site records storage, training services, etc.).

1.7.2.2.3

This QAPD, organized to present the QA Program for the Cook Nuclear Plant in the order of the 1 criteria of 10CFR50, Appendix B, states AEPSC policy for each of the criteria and describes how the controls

pertinent to each are carried out. Any changes made to this QAPD that do not reduce the commitments previously accepted by the NRC must be submitted to the NRC at least annually. Any changes made to this QAPD that do reduce the commitments previously accepted by the NRC must be submitted to the NRC and receive NRC approval prior to implementation. The submittal of the changes described above shall be made in accordance with the requirements of 10CFR50.54.

The program described in this QAPD will not be intentionally changed in any way that would prevent it from meeting the criteria of 10CFR50, Appendix B and other applicable operating license requirements.

1.7.2.2.4

Documents used for implementing the provisions of this QAPD include the following:

Plant Manager Instructions (PMIs) establish the policy at the plant for compliance with specified criteria, and assign responsibility to the various departments, as required, for implementation. Plant Manager Procedures (PMPs), Department Head Procedures (DHPs), and in some cases Department Head Instructions (DHIs), have been prepared to describe the detailed activities required to support safe and effective plant operation as per the PMIs.

The PMIs are reviewed by AEPSC QA for concurrence that they will satisfactorily implement regulatory requirements and commitments. PMIs and PMPs are reviewed by the PNSRC prior to approval by the Plant Manager.

DHPs and DHIs are reviewed within the departments prior to approval by the Department Head of origination. DHPs and DHIs that might involve an unreviewed safety question as defined in 10CFR50.59 are reviewed by PNSRC prior to approval by the Department Head of origination.

AEPSC General Procedures (GPs) are utilized to define corporate policies and requirements for quality assurance, and to implement certain corporate QA Program requirements. AEPSC division/department and/or section procedures are also used to implement QA Program requirements.

GPs may also be used to define policies which are nonprocedural in nature.

[illegible]

When contractors perform work on-site under their own quality assurance programs, the programs are audited for compliance and consistency with the applicable requirements of the Cook Nuclear Plant's QA Program and the contract, and are approved by AEPSC QA prior to the start of work. Implementation of on-site contractor's QA programs, will be audited to assure that the contractor's programs are effective.

1.7.2.2.5

Provisions of the QA Program for the Cook Nuclear Plant apply to activities affecting the quality of safety-related items. Appendix A to this QAPD lists the Regulatory/Safety Guides and ANSI Standards that identify AEPSC's commitment. Appendix B describes necessary exceptions and clarifications to the requirements of those documents. The scope of the program, and the extent to which its controls are applied, are established as follows:

a) AEPSC uses the criteria specified in the Cook Nuclear Plant Updated FSAR for identifying structures, systems and components to which the QA Program applies.

b) This identification process results in the Facility Data Base for the Cook Nuclear Plant. This Facility Data Base is controlled by authorized personnel. Facility Data Base items are determined by

engineering analysis of the function(s) of plant items in relation to safe operation and shutdown.

c) The extent to which controls specified in the QA Program are applied to Facility Data Base items is determined for each item considering its relative importance to safety. Such determinations are based on data in such documents as the Cook Nuclear Plant Technical Specifications and the Updated FSAR.

1.7.2.2.6

Activities affecting safety-related items are accomplished under controlled conditions. Preparations for such activities include consideration of the following:

a) Assigned personnel are qualified.
b) Work has been planned to applicable engineering and/or Technical Specifications.
c) Specified equipment and/or tools are available.
d) Items are in an acceptable status.
e) Items on which work is to be performed are in the proper condition for the task.
f) Proper approved instructions/procedures for the work are available for use.
g) Items and facilities that could be damaged by the work have been protected, as required.
h) Provisions have been made for special controls, processes, tests and verification methods.

1.7.2.2.7

Responsibility and authority for planning and implementing indoctrination and training of AEPSC and Cook Nuclear Plant staff personnel are specifically designated, as follows:

a) The training and indoctrination program provides for on-going training and periodic familiarization with the QA Program for the Cook Nuclear Plant.

b) Personnel who perform inspection and examination functions are qualified in accordance with requirements of Regulatory Guide 1.8, ANSI N18.1, Regulatory Guide 1.58, ANSI N45.2.6, the ASME B&PV Code, or SNT-TC-1A, as applicable, and with exceptions as noted in Appendix B hereto.

c) AEPSC QAD auditors are qualified in accordance with Regulatory Guide 1.146 and ANSI N45.2.23.

d) Personnel assigned duties such as special cleaning processes, welding, etc., are qualified in accordance with applicable codes, standards, regulatory guides and/or plant procedures.

e) The training, qualification and certification program includes, as applicable, provisions for retraining, reexamination and recertification to ensure that proficiency is maintained.

f) Training, qualification, and certification records including documentation of objectives, waivers/exceptions, attendees and dates of attendance, are maintained at least as long as the personnel involved are performing activities to which the training, qualification and certification is relevant.

g) Personnel responsible for performing activities that affect safety-related items are instructed as to the purpose, scope and implementation of the applicable manuals, instructions and procedures.

Management/supervisory personnel receive functional training to the level necessary to plan, coordinate and administer the day-to-day

verification activities of the QA Program for which they are responsible.

Training of AEPSC and Cook Nuclear Plant personnel is performed employing the following techniques, as applicable: 1) on the job and formal training administered by the department or section the individual works for; 2) formal training conducted by qualified instructors from the Cook Nuclear Plant Training Department or other entities (internal and external to the AEP System); and 3) formal, INPO accredited training conducted by the Cook Nuclear Plant Training Department. Records of training sessions for such training are maintained. Where personnel qualifications or certifications are required, these certifications are performed on a scheduled basis (consistent with the appropriate code or standard).

Cook Nuclear Plant employees receive introductory training in quality assurance usually within the first two weeks of employment. In addition, AEPSC personnel receive training prior to being allowed unescorted access to the plant. This training includes management's policy for implementation of the QA Program through Plant Manager and Department Head Instructions and Procedures. These instructions also include a description of the QA Program, the use of instructions and procedures, personnel requirements for procedure compliance and the systems and components controlled by the QA Program.

1.7.3 DESIGN CONTROL

1.7.3.1 SCOPE

Design changes are accomplished in accordance with approved design. Activities to develop such designs are controlled. Depending on the type of design change, these activities include design and field engineering; the performance of physics, seismic, stress, thermal, hydraulic and radiation evaluations; update of the FSAR; review of accident analyses; the development and control of associated computer programs; studies of material compatibility; accessibility for inservice

inspection and maintenance; determination of quality standards; and requirement for equipment qualification. The controls apply to preparation and review of design documents, including the correct translation of applicable regulatory requirements and design bases into design, procurement and procedural documents.

1.7.3.2 IMPLEMENTATION

1.7.3.2.1

Design changes are controlled by procedures and instructions and are reviewed as required by 10CFR50.59 and the Technical Specifications.

Safety-related design changes are accomplished by one of two separate processes: Minor Modification (MM), or Request for Change (RFC). Those that do not alter the intended function of the item and can be determined by judgement to have a minimal overall impact on the item being modified may be implemented via the MM process. All other safety-related design changes, that are not appropriate for MM processing, are implemented via the RFC process.

In cases where design changes could be deemed to be within the scope of RFCs or MMs solely due to possible insignificant adverse seismic effects, the change may be implemented via the Plant Modification (PM) process.

In the case where safety-related items are involved and the change introduces only insignificant adverse seismic effects, the change may be implemented via the Plant Modification (PM) process.

1.7.3.2.2

Design changes are reviewed to determine their impact on nuclear safety and to determine if the proposed changes involve an unreviewed safety question as defined by 10CFR50.59. If a design change were to involve

an unreviewed safety question, it would not be approved for implementation until the required NRC approval was received.

RFCs (except those requiring emergency processing), MMs and PMs (having only insignificant seismic effect on safety items) are reviewed and approved prior to implementation, as a minimum, by the cognizant AEPSC section and Plant Manager. The PNSRC also reviews those RFCs, MMs, and PMs for which safety evaluations are deemed necessary, pursuant to 10CFR50.59 and Technical Specification 6.5.1.6.

1.7.3.2.3

For RFCs, the Change Control Board established within AEPSC provides an additional review and approval level. The Change Control Board is comprised of members of the Engineering, Design, Nuclear Operations and QA organizations within AEPSC, and is supplemented by other AEPSC organizations or individuals, as required.

The cognizant member of the Change Control Board assigns a lead engineer for each RFC. The lead engineer is responsible for coordinating the RFC activities within AEPSC and maintaining close interface with AEPSC Site Engineering Support Project Engineering.

1.7.3.2.4

Proposed RFCs which require emergency processing are originated at the plant, reviewed by the PNSRC, and approved by the Plant Manager. Cook Nuclear Plant management then contacts the AEPSC NOD, and other AEPSC management, as required, describes the change requested, and implements the change only after receiving verbal AEPSC management authorization to proceed. These reviews and approvals are documented and become a part of the RFC Packet.

1.7.3.2.5

When RFCs or MMs involve design interfaces between internal or external design organizations, or across technical disciplines, these interfaces are controlled. Procedures are used for the review, approval, release, distribution and revision of documents involving design interfaces to ensure that structures, systems and components are compatible geometrically and functionally with processes and the environment. Lines of communication are established for controlling the flow of needed design information across design interfaces, including changes to the information as work progresses. Decisions and problem resolutions involving design interfaces are made by the AEPSC organization having responsibility for engineering direction of the design effort.

1.7.3.2.6

Checks are performed and documented to verify the dimensional accuracy and completeness of design drawings and specifications.

1.7.3.2.7

RFC design document packages are audited by AEPSC QA to assure that the documents have been prepared, verified, reviewed and approved in accordance with company procedures.

1.7.3.2.8

The extent of, and methods for, design verification are documented. The extent of design verification performed is a function of the importance of the item to safety, design complexity, degree of standardization, the state-of-the-art, and similarity with previously proven designs. Methods for design verification include evaluation of the applicability of standardized or previously proven designs, alternate calculations, qualification testing and design reviews. These methods may be used

singly or in combination, depending on the needs for the design under consideration.

When design verification is done by evaluating standardized or previously proven designs, the applicability of such designs is confirmed. Any differences from the proven design are documented and evaluated for the intended application.

Qualification testing of prototypes, components, or features is used when the ability of an item to perform an essential safety function cannot otherwise be adequately substantiated. This testing is performed before plant equipment installation, where possible, but always before reliance upon the item to perform a safety-related function. Qualification testing is performed under conditions that simulate the most adverse design conditions, considering all relevant operating modes. Test requirements, procedures and results are documented. Results are evaluated to assure that test requirements have been satisfied. Design changes shown to be necessary through testing are made, and any necessary retesting or other verification is performed. Test configurations are clearly documented.

Design reviews are performed by multi-organizational or interdisciplinary groups, or by single individuals. Criteria are established to determine when a formal group review is required, and when review by an individual is sufficient.

Procedures require that minor design changes accomplished by the MM process also receive formal design verification. Applicable design verification activities shall be completed prior to declaring the design change, or portion thereof, operational.

1.7.3.2.9

Persons representing applicable technical disciplines are assigned to perform design verifications. These persons are qualified by

appropriate education or experience, but are not directly responsible for the design. The designer's immediate supervisor may perform the verification, provided that:

1) The supervisor is the only technically qualified individual.

<u>or</u>

2) The supervisor has not specified a singular design approach, ruled out design considerations, nor established the design inputs.

<u>and</u>

3) The need is individually documented and approved in advance by the supervisor's management.

<u>and</u>

4) Regularly scheduled QA audits verify conformance to previous items 1 through 3.

Design verification on safety-related design changes shall be completed prior to declaring a design change, or portions thereof, operational.

1.7.3.2.10

Implementation of design changes is coordinated on site by AEPSC Site Engineering Support Project Engineering. Material to perform the design change must meet the specifications established for the original system, or as specified by the lead engineer. For those design changes where testing after completion is required, the testing documentation is reviewed by the organization performing the test and, when specified, by the AEPSC lead engineer or other cognizant engineer(s). Further, completed design changes are audited/surveilled by AEPSC QA following installation and testing.

1.7.3.2.11

Changes to design documents, including field changes, are reviewed, approved and controlled in a manner commensurate with that used for the

original design. Such changes are evaluated for impact. Information on approved changes is transmitted to all affected organizations.

1.7.3.2.12

Error and deficiencies in, and deviations from, approved design documents are identified and dispositioned in accordance with established design control and/or corrective action procedures.

1.7.3.2.13

Established design control procedures provide for:

1) controlled submission of design changes,
2) engineering evaluation,
3) review for impact on nuclear safety,
4) audit by AEPSC QA,
5) design modification,
6) AEPSC managerial review, and
7) approval and record keeping for the implemented design change.

1.7.4 PROCUREMENT DOCUMENT CONTROL

1.7.4.1 SCOPE

Procurement documents define the characteristics of item(s) to be procured, identify applicable regulatory and industry codes/standards requirements, and specify supplier QA Program requirements to the extent necessary to assure adequate quality.

1.7.4.2 IMPLEMENTATION

1.7.4.2.1

Procurement control is established by instructions and procedures. These documents require that purchase documents be sufficiently detailed to ensure that purchased materials, components and services associated

with safety-related structures or systems are: 1) purchased to specification and code requirements equivalent to those of the original equipment or service (except when the Code of Federal Regulations requires upgrading), 2) properly documented to show compliance with the applicable specifications, codes and standards, and 3) purchased from vendors or contractors who have been evaluated and deemed qualified, or by the commercial grade dedication process.

Procedures establish the review of procurement documents to determine that: quality requirements are correctly stated, inspectable and controllable; there are adequate acceptance criteria; and procurement documents have been prepared, reviewed and approved in accordance with established requirements.

The manager of the originating group, with support of the cognizant AEPSC engineering group, is responsible for assuring that applicable requirements are set forth in procurement documents.

The Cook Nuclear Plant may request assistance of AEPSC cognizant engineers in any procurement activity.

1.7.4.2.2

The Facility Data Base, in conjunction with other sources, is used for equipment safety classification and procurement grade. AEPSC specifications are used to determine requirements, codes or standards that items must fulfill, and define the documentation that must accompany the item to the plant.

Procurement documents for safety-related items and services are reviewed to ensure that: correct classification is made; the requirements are properly stated; and that measures have been, or will be, implemented to assure the requirements are met and adequately provided for.

Purchase requisitions for new safety-related items are initiated by the cognizant engineering group which establishes initial requirements.

Replacement/spares are purchased to requirements equivalent to the original unless upgrading is required by Federal Regulations, or deemed necessary by the cognizant engineering group.

1.7.4.2.3

The contents of procurement documents vary according to the item(s) being purchased and its function(s) in the Cook Nuclear Plant. Provisions of this QAPD are considered for application to service contractors, also. As applicable, procurement documents include:

a) Scope of work to be performed.
b) Technical requirements, with applicable drawings, specifications, codes and standards identified by title, document number, revision and date, with any required procedures, such as special process instructions identified in such a way as to indicate source and need. Imposition of guides/standards on AEPSC/I&M suppliers and subtier suppliers will be on a case-by-case basis depending upon the item or service to be supplied and upon the degree that AEPSC/I&M relies on suppliers to invoke guides/standards. AEPSC/I&M recognizes that certain suppliers have acceptable 10CFR50, Appendix B QA programs, even through, the suppliers are not committed to Regulatory Guides or industry standards (e.g. ANSI N45.2.6.). In those cases, in which suppliers are not committed to the same guides/standards as AEPSC/I&M, AEPSC/I&M will assure that (1) the supplier's QA program provides adequate QA controls, regardless of the lack of specific commitment, or (2) controls will be invoked directly by AEPSC/I&M to assure adequate quality of items/services received by suppliers.
c) Regulatory, administrative and reporting requirements.
d) Quality requirements appropriate to the complexity and scope of the work, including necessary tests and/or inspections.

e) A requirement for a documented QA Program, subject to QA review and written concurrence prior to the start of work.

f) A requirement for the supplier to invoke applicable quality requirements on subtier suppliers.

g) Provisions for access to supplier, and subtier suppliers', facilities and records for inspections, surveillances and audits.

h) Identification of documentation to be provided by the supplier, the schedule of submittals and documents requiring AEPSC approval.

1.7.4.2.4

The AEPSC QA Division performs audits of procurement documents to assure that QA Program requirements have been met. These audits are conducted in accordance with AEPSC QA·Division procedures.

1.7.4.2.5

Changes to procurement documents are controlled in a manner commensurate with that used for the original documents.

1.7.5 INSTRUCTIONS, PROCEDURES, AND DRAWINGS

1.7.5.1 SCOPE

Activities affecting the quality of safety-related structures, systems and components are accomplished using instructions, procedures and drawings appropriate to the circumstances, including acceptance criteria for determining if an activity has been satisfactorily completed.

1.7.5.2 IMPLEMENTATION

1.7.5.2.1

Instructions and procedures incorporate: 1) a description of the activity to be accomplished, and 2) appropriate quantitative (such as tolerances and operating limits) and qualitative (such as workmanship and standards) acceptance criteria sufficient to determine that the

The PMIs have been classified into the following series:

1000 Personnel Selection, PNSRC Procedures

2000 Administration - Document Control, Security, Training, Records, Emergency Plan, Fire Protection, Clearance Permits, Chemical Control, Internal Cleanliness, Spill Response, Standing orders, Corrective Maintenance.

3000 Procurement, Receiving, Shipping and Storage

4000 Operations, Fuel Handling, Surveillance Testing, Test Controls

5000 Maintenance, Repair, Modification, Special Processes, EQ and ISI Control of Contractors

6000 Technical - Chemistry/Radiological Controls, Radiation Protection, Performance/Engineering Testing, and Instrument and Control Maintenance and Calibration, Measuring and Test Equipment

7000 Quality Assurance, Quality Control Program and Condition/Problem Reporting

Instructions and procedures identify the regulatory requirements and commitments which pertain to the subject that it will control and establish responsibilities for implementation. Instructions and procedures may either provide the guidance necessary for the development of supplemental instructions and/or procedures to implement their requirements, or provide comprehensive guidance based on the subject matter.

1.7.5.2.4

Cook Nuclear Plant drawings are produced, controlled and distributed under the control of AEPSC and the Cook Nuclear Plant. AEPSC design drawings are produced by, or under the control of, the AEPSC Nuclear Engineering Department under a set of procedures which direct their development and review. These procedures specify requirements for inclusion of quantitative and qualitative acceptance criteria. Specific

activity has been satisfactorily accomplished. Hold points for inspection are established when required.

Instructions and procedures pertaining to the specification of, and/or implementation of, the QA Program receive multiple reviews for technical adequacy and inclusion of appropriate quality requirements. Top tier instructions and procedures are reviewed and/or approved by AEPSC QA. Lower tier documents are reviewed and approved, as a minimum, by management/supervisory personnel trained to the level necessary to plan, coordinate and administer those day-to-day verification activities of the QA Program for which they are responsible.

Special procedures may be issued for activities which have short-term applicability.

1.7.5.2.2

AEPSC activities relative to the Cook Nuclear Plant are outlined by procedures which provide the controls for the implementation of these activities. AEPSC has two categories of QA Program implementation procedures:

1) General Procedures (GPs) which are applicable to all AEPSC divisions and departments involved with Cook Nuclear Plant.

2) Organization procedures which apply to the specific division, department or section involved.

1.7.5.2.3

Activities at the Cook Nuclear Plant are controlled using plant procedures.

drawings are reviewed and approved by the cognizant engineering organization.

AEPSC has stationed an on-site design staff to provide for the revision of certain types of design drawings to reflect as-built conditions.

1.7.5.2.5

Complex Cook Nuclear Plant procedures are designated as "In Hand" procedures. Examples of "In Hand" procedures are those developed for extensive or complex jobs where reliance on memory cannot be trusted. Further, those procedures which describe a sequence which cannot be altered, or require the documentation of data during the course of the procedure, are considered. "In Hand" procedures are designated as such by double asterisks (**) which precede the procedure number on the cover sheet, all pages and attachments of a procedure and the corresponding index.

1.7.6 DOCUMENT CONTROL

1.7.6.1 SCOPE

Documents controlling activities within the scope defined in 1.7.2 herein are issued and changed according to established procedures. Documents such as instructions, procedures and drawings, including changes thereto, are reviewed for adequacy, approved for release by authorized personnel, and are distributed and used at the location where a prescribed activity is performed.

Changes to controlled documents are reviewed and approved by the same organizations that performed the original review and approval, or by other qualified, responsible organizations specifically designated in accordance with the procedures governing these documents. Obsolete or superseded documents are controlled to prevent inadvertent use.

1.7.6.2 IMPLEMENTATION

1.7.6.2.1

Controls are established for approval, issue and change of documents in the following categories:

a) Design documents (e.g., calculations, specifications, analyses)
b) Drawings and related documents
c) Procurement documents
d) Instructions and procedures
e) Updated Final Safety Analysis Report (UFSAR)
f) Plant Technical Specifications
g) Safeguards documents

1.7.6.2.2

The review, approval, issuance and change of documents are controlled by:

a) Establishment of criteria to ensure that adequate technical and quality requirements are incorporated.

b) Identification of the organization responsible for review, approval, issue and maintenance.

c) Review of changes to documents by the organization that performed the initial review and approval, or by the organization designated in accordance with the procedure governing the review and approval of specific types of documents.

[redacted]

1.7.6.2.3

Documents are issued and controlled so that:

a) The documents are available prior to commencing work.
b) Obsolete documents are replaced by current documents in a timely manner.

1.7.6.2.4

Master lists, or equivalent controls, are used to identify the current revision of instructions, procedures, specifications and drawings. These control documents are updated and distributed to designated personnel who are responsible for maintaining current copies of the applicable documents. The distribution of controlled documents is performed under procedures requiring receipt acknowledgement and in accordance with established distribution lists.

1.7.6.2.5

In the event a drawing is developed on-site to reflect an as-built configuration, the marked-up drawing is maintained in the Master Plant File and all holders of the drawing are issued appropriate notification to inform them the revision they hold is not current, cannot be used and, if required, reference must be made to the Master Plant File drawing.

1.7.6.2.6

Documents prepared for use in training are appropriately marked to indicate that they cannot be used to operate or maintain the facility or to conduct activities affecting the quality of safety-related items. At the Cook Nuclear Plant, unless a document is identified as 'controlled' or 'working copy' only, it is automatically assumed that the document is for information use only.

1.7.7 CONTROL OF PURCHASED ITEMS AND SERVICES

1.7.7.1 SCOPE

Activities that implement approved procurement requests for items and services are controlled to assure conformance with procurement document requirements. Controls include a system of supplier evaluation and selection audits, acceptance of items and documentation upon delivery, and periodic assessment of supplier performance. Objective evidence of quality that demonstrates conformance with specified procurement document requirements is available to the Cook Nuclear Plant site prior to use of equipment, material, or services.

1.7.7.2 IMPLEMENTATION

1.7.7.2.1

AEPSC qualifies suppliers and distributors by performing a documented evaluation of their capability to provide items or services specified by procurement documents. Items and services designated as safety-related are purchased from suppliers whose QA programs have been accepted in accordance with AEPSC requirements, or from commercial grade suppliers through the AEPSC dedication program. Suppliers of other items/services are subject to evaluation and approval based on acceptance criteria applicable to those items/services.

Qualification of such suppliers is determined by the AEPSC QA Division. In the discharge of this responsibility, the AEPSC QA Division may use information generated by other utilities. The supplier, or distributor, must be approved before procurement can be completed. AEPSC is a member of the Nuclear Procurement Issues Committee (NUPIC), participates in joint supplier audits, and shares audit information consistent with NUPIC requirements. The supplier, or distributor, must be acceptable, or acceptable subject to follow-up, before a procurement can be approve and processed. Additional audits will be conducted, as necessary, to meet requirements. Acceptance is not complete until it has been

determined that the suppliers' QA program can meet the requirements for the item(s)/service(s) offered.

1.7.7.2.2

For items that are not unique to a nuclear power plant ("Commercial Grade") where requirements cannot be imposed in a practical manner at the time of procurement, programs for dedication to safety-related standards are established and accomplished by the AEPSC cognizant engineer prior to the item being accepted for safety-related use.

1.7.7.2.3

In-process audits of suppliers' activities during fabrication, inspection, testing and shipment of items are performed when deemed necessary, depending upon supplier qualification status, complexity of the item(s) being furnished, the items' importance to safety, and/or previous supplier history. These audits are performed by AEPSC QA. The cognizant engineer and/or responsible Cook Nuclear Plant personnel may also participate, if deemed necessary.

1.7.7.2.4

Spare and replacement parts are procured in such a manner that their performance and quality are at least equivalent to those of the parts that will be replaced.

a) Specifications and codes referenced in procurement documents for spare or replacement items are at least equivalent to those for the original items or to properly reviewed and approved revisions.

b) Parts intended as spares or replacement for "off-the-shelf" items, or other items for which quality requirements were not originally specified, are evaluated for performance at least equivalent to the original.

c) Where quality requirements for the original items cannot be determined, requirements and controls are established by engineering evaluation performed by qualified individuals. The evaluation assures there is no adverse effect on interfaces, safety, interchangeability, fit, form, function, or compliance with applicable regulatory or code requirements. Evaluation results are documented.

d) Any additional or modified design criteria, imposed after previous procurement of the item(s), are identified and incorporated.

1.7.7.2.5

Instructions and procedures address requirements for supplier selection and control, as well as procurement document control. The PMI on receipt inspection of safety-related items addresses the program for inspection of incoming items, including a review of the documentation required under the procurement. Receipt inspection personnel are qualified and certified in accordance with the requirements of ANSI N45.2.6. Provisions for receipt inspection apply regardless of where the procurement originates. Additional inspections may apply if required by the procurement document.

Where items and/or services are safety-related and procurement is accomplished without assistance of AEPSC, supplier selection is limited to those companies identified as being qualified.

1.7.7.2.6

Items received at the site are tagged with a "HOLD" tag and/or placed in a designated area (e.g. new fuel) until receipt inspected. During receipt inspection, designated material characteristics and attributes are checked, and documentation is checked against the procurement documents. If found acceptable, the "HOLD" tag is removed and replaced with an "ACCEPTED" tag and the item is placed in a designated area of

the storeroom. Item traceability to procurement documents and to end use is maintained through recording of "HOLD" and "ACCEPTED" tag numbers on applicable documents.

Nonconforming items, or missing or questionable documentation results in items being placed on "HOLD" and maintained in a designated, controlled area of the storeroom. If the nonconformance cannot be cleared, the item is either scrapped, returned to manufacturer, or dispositioned through engineering analysis.

1.7.7.2.7

Contractors providing services (on-site) for safety-related components are required to have either a formal quality assurance program and procedures, or they must abide by the Cook Nuclear Plant QA Program and procedures. Prior to their working at the Cook Nuclear Plant, contractor quality assurance programs must be audited and approved by AEPSC QA. Contractor procedures must be reviewed and approved by the originating/sponsoring department head. Further, periodic audits of site contractor activities are conducted under the direction of the AEPSC Quality Assurance Superintendent.

1.7.7.2.8

To the extent prescribed in specific procurement documents, suppliers furnish quality records; documentary evidence that material and equipment either conforms to requirements or identifies any requirements that have not been met; and descriptions of those nonconformances from the procurement requirements, which have been dispositioned "use-as-is" or "repair." This evidence is retained at the Plant, or at the Service Corporation.

To the extent prescribed in specific procurement agreements, suppliers are required to maintain additional (backup) documents in their record system.

In some cases, such as with NSSS, suppliers are designated primary record retention responsibility.

1.7.7.2.9

The capability of suppliers to furnish valid documentation is evaluated during procurement document reviews, annual supplier evaluations, and during audits.

1.7.8 IDENTIFICATION AND CONTROL OF ITEMS

1.7.8.1 SCOPE

Items are identified and controlled to prevent their inadvertent use. Identification of items is maintained either on the items, their storage areas or containers, or on records traceable to the items.

1.7.8.2 IMPLEMENTATION

1.7.8.2.1

Controls are established that provide for the identification and control of items (including partially fabricated assemblies).

1.7.8.2.2

Items are identified by physically marking the item or its container, and by maintaining records traceable to the item. The method of identification is such that the quality of the item is not degraded.

1.7.8.2.3

Items are traceable to applicable drawings, specifications, or other pertinent documents to ensure that only correct and acceptable items are used. Verification of traceability is performed and documented prior to release for fabrication, assembly, or installation.

1.7.8.2.4

Requirements for the identification by use of heat number, part number, serial number, etc., are included in AEPSC Specifications (DCCs) and/or the procurement document.

1.7.8.2.5

Separate storage is provided for incorrect or defective items that are on hold and material which has been accepted for use. All safety-related items are appropriately tagged or identified (stamping, etc.) to provide easy identification as to the items' usage status. Records are maintained for the issue of items to provide traceability from storage to end use in the Cook Nuclear Plant.

1.7.8.2.6

When materials are subdivided, appropriate identification numbers are transferred to each section of the material, or traceability is maintained through documentation.

1.7.9 CONTROL OF SPECIAL PROCESSES

1.7.9.1 SCOPE

Special processes are controlled and accomplished by qualified personnel using approved procedures and equipment in accordance with applicable codes, standards, specifications, criteria and other special requirements.

1.7.9.2 IMPLEMENTATION

1.7.9.2.1

Processes subject to special process controls are those for which full verification or characterization by direct inspection is impossible or impractical. Such processes include welding, heat treating, chemical

cleaning, application of protective coatings, concrete placement and NDE.

1.7.9.2.2

Special process requirements for chemical cleaning, application of protective coatings and concrete placement are set forth in AEPSC Specifications (DCCs) and/or directives prepared by the responsible AEPSC cognizant engineer. These documents are reviewed and approved by other personnel with the necessary technical competence. AEPSC Specifications are audited by the AEPSC QA Division.

Special process requirements for welding, heat treating and NDE are set forth in AEPSC Specifications, the AEP Welding and NDE Manuals and plant procedures. These specifications and manuals are prepared by, or are reviewed and approved by, the AEPSC Cognizant Engineer - Welding and NDE Administrator. The administrative controls portion of the NDE Manual is audited by AEPSC QA.

Special process procedures, with the exception of welding and heat treating, are prepared by Cook Nuclear Plant personnel with technical knowledge in the discipline involved. These procedures, which are also reviewed by other personnel with the necessary technical competence, are qualified by testing.

Welding is performed in accordance with procedures contained in the AEP Welding Manual, or by approved contractor's procedures. These procedures are qualified in accordance with applicable codes, and Procedure Qualification Records are prepared. Weld Procedure Qualification Records are reviewed and approved by the AEPSC Cognizant Engineer - Welding. Weld qualification documentation is retained in the AEP Welding Manual, or the approved contractor's manual.

Contractor welding procedures are qualified by the contractor. These procedures and the qualification documentation are reviewed and approved

by the AEPSC Cognizant Engineer - Welding. This documentation is retained by the contractor.

1.7.9.2.3

NDE personnel are qualified and certified by a Cook Nuclear Plant NDE Level III who has been qualified and certified by the designated AEPSC NDE Administrator. Certification is by examination. Personnel qualification is kept current by re-examination at time intervals specified in qualification/certification procedures which are in accordance with the ASME Code.

Cook Nuclear Plant welders are qualified by the Maintenance Department using AEPSC approved procedures. Supervision of Cook Nuclear Plant welder qualifications is performed by the Maintenance Department. Examination of specimens is performed under the supervision of the Safety and Assessment Department in accordance with the AEP Welding Manual covering welder qualification. Cook Nuclear Plant welder qualification records are maintained for each welder by the Maintenance Department. Contractor and craft welders are qualified by the contractor using procedures approved by the AEPSC Cognizant Engineer - Welding in accordance with AEPSC procedures. Contractor and craft welder qualification records are maintained by the contractor.

1.7.9.2.4

QC/NDE Technicians assigned to the Safety and Assessment Department perform nondestructive testing for work performed by Cook Nuclear Plant and contractor personnel. These individuals are qualified to either SNT-TC-1A, or ANSI N45.2.6, and records of the qualifications/certifications are maintained at Cook Nuclear Plant.

1.7.9.2.5

For special processes that require qualified equipment, such equipment is qualified in accordance with applicable codes, standards and specifications.

1.7.9.2.6

Special process qualifications are reviewed during regularly scheduled QA audits. Qualification records are maintained in accordance with 1.7.17 herein.

1.7.9.2.7

The documentation resulting from welding and nondestructive testing is reviewed by appropriate personnel.

1.7.10 INSPECTION

1.7.10.1 SCOPE

Activities affecting the quality of safety-related structures, systems and components are inspected to verify their conformance with requirements. These inspections are performed by personnel other than those who perform the activity. Inspections are performed by qualified personnel utilizing written procedures which establish prerequisites and provide documentation for evaluating test and inspection results. Direct inspection, process monitoring, or both, are used as necessary. When applicable, hold points are used to ensure that inspections are accomplished at the correct points in the sequence of activities.

1.7.10.2 IMPLEMENTATION

1.7.10.2.1

Inspections are applied to appropriate activities to assure conformance to specified requirements.

Hold points are provided in the sequence of procedures to allow for the inspection, witnessing, examination, measurement, or review necessary to assure that the critical, or irreversible, elements of an activity are

e) Identification of personnel responsible for performing the inspection.

f) Acceptance and rejection criteria.

g) Recording of the inspection results and the identification of the inspector.

1.7.10.2.4

[illegible]

Inspections are conducted using the following programs:

a. Peer Inspection Program. The Peer Inspection Program is based on the premise that I&M personnel are qualified to ANSI N18.1 (1971), Selection and Training of Nuclear Power Plant Personnel, and are periodically trained in their skill area using INPO accredited training. As a result of their experience, qualifications, and training, I&M personnel may perform inspections of work functions associated with normal operation of the Plant, routine maintenance, and certain routine technical activities which are routinely performed by I&M personnel (peers). Peer inspection personnel are independent in that they do not perform or directly supervise the work being inspected, but they may be from the same work group. D. C. Cook Plant Safety and Assessment QC/NDE personnel, qualified in accordance with Regulatory Guide 1.58 and ANSI N45.2.6, will ensure (via surveillance) that peer inspections are being correctly implemented and make periodic reports to management.

b. ANSI N45.2.6 Inspection Program. Major modification and non-routine maintenance work on safety related equipment is inspected per ANSI N45.2.6, Qualification of Inspection, Examination, and Testing Personnel for Nuclear Power Plants, whether it is

being performed as required. Note that hold points may not apply to all procedures, but each must be reviewed for this attribute.

Hold points specify exactly what is to be done (e.g., type of inspection or examination, etc.), acceptance criteria, or reference to another procedure, etc., for the satisfactory completion of the hold point. When included in the sequence of a procedure, the activities required by hold points are completed prior to continuing work beyond that point.

Process monitoring is used in whole, or in part, where direct inspection alone is impractical or inadequate.

1.7.10.2.2

Training, qualification and certification programs for personnel who perform inspections are established, implemented and documented in accordance with 1.7.2 herein and as described in Appendix B hereto, item 9b, with exceptions as noted therein.

1.7.10.2.3

Inspection requirements are specified in procedures, instructions, drawings, or checklists as applicable. They provide for the following, as appropriate:

a) Identification of applicable revisions of required instructions,drawings and specifications.

b) Identification of characteristics and activities to be inspected.

c) Inspection methods.

d) Specification of measuring and test equipment having the necessary accuracy.

performed by I&M or contractor personnel. All safety related work performed by contract personnel is inspected per ANSI N45.2.6. Inspections of these work activities are performed by inspectors qualified and certified in accordance with Regulatory Guide 1.58 and ANSI N45.2.6. Contractors performing work on safety related equipment are required to comply with the applicable requirements of Regulatory Guide 1.33 and ANSI N45.2.

1.7.10.2.5

. Inspections associated with the packaging and shipment of radioactive waste and materials are conducted using the following program:

a) <u>NRC Licensed Packagings</u> - Inspections of NRC licensed radioactive material packagings shall be performed by individuals independent from the work being performed. The independent inspectors shall be Indiana Michigan Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum. Additionally, the inspector shall be familiar with the activities being performed.

b) <u>Non-NRC Licensed Packagings and Containers</u> - Inspections of non-NRC licensed radioactive material packagings and containers (shipping and/or burial) shall be performed by Indiana Michigan Power personnel, qualified in accordance with Regulatory Guide 18. and ANSI N18.1, as a minimum.

c) <u>Transportation Vehicles</u> - Inspection of transportation vehicles being shipped as "exclusive use", shall be performed by Indiana Michigan Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum.

d) <u>Other inspections and Verification</u> - Inspections and verifications of other activities associated with the packaging and shipment of radioactive materials and waste shall be performed by Indiana and

Michigan Power personnel, qualified in accordance with Regulatory Guide 1.8 and ANSI N18.1, as a minimum.

1.7.10.2.6

Inspections are performed, documented, and the results evaluated by designated personnel in order to ensure that the results substantiate the acceptability of the item or work. Evaluation and review results are documented.

1.7.11 TEST CONTROL

1.7.11.1 SCOPE

Testing is performed in accordance with established programs to demonstrate that structures, systems and components will perform satisfactorily in service. The testing is performed by qualified personnel in accordance with written procedures that incorporate specified requirements and acceptance criteria. Types of tests are:

Scheduled

Surveillance, preventive maintenance, post-design, qualification.

Unscheduled

Pre- and post-maintenance.

Test parameters (including any prerequisites), instrumentation requirements, and environmental conditions are specified in test procedures. Test results are documented and evaluated.

1.7.11.2 IMPLEMENTATION

1.7.11.2.1

Tests are performed in accordance with programs, procedures and criteria that designate when tests are required and how they are to be performed. Such testing includes the following:

a) Qualification tests, as applicable, to verify design adequacy.

b) Acceptance tests of equipment and components to assure their operation prior to delivery or installation.

c) Post-design tests to assure proper and safe operation of systems and equipment prior to unrestricted operation.

d) Surveillance tests to assure continuing proper and safe operation of systems and equipment. The PMI on surveillance testing controls the periodic testing of equipment and systems to fulfill the surveillance requirements established by the Technical Specifications. Controls have been established to identify uncompleted surveillance testing to assure it is rescheduled for completion to meet Technical Specification frequency requirements. Data taken during surveillance testing is reviewed by appropriate management personnel to assure that acceptance criteria is fulfilled, or corrective action is taken to correct deficiencies.

e) Maintenance tests after preventive or corrective maintenance.

1.7.11.2.2

Test procedures, as required, provide mandatory hold points for witness or review.

1.7.11.2.3

Testing is accomplished after installation, maintenance, or repair, by surveillance test procedures, or performance tests, which must be satisfactorily completed prior to determining the equipment is in an operable status. All data resulting from these tests is retained at the Cook Nuclear Plant after review by appropriate management personnel.

1.7.12 CONTROL OF MEASURING AND TEST EQUIPMENT

1.7.12.1 SCOPE

Measuring and testing equipment used in activities affecting the quality of safety-related systems, components and structures are properly identified, controlled, calibrated and adjusted at specified intervals to maintain accuracy within necessary limits.

1.7.12.2 IMPLEMENTATION

1.7.12.2.1

Established procedures and instructions are used for calibration and control of measuring and test equipment utilized in the measurement, inspection and monitoring of structures, systems and components. These procedures and instructions describe calibration techniques and frequencies, and maintenance and control of the equipment.

AEPSC QA periodically assesses the effectiveness of the calibration program via the QA audit program.

1.7.12.2.2

Measuring and test equipment is uniquely identified and is traceable to its calibration source.

1.7.12.2.3

A system has been established for attaching, or affixing labels, to measuring and test equipment to display the date calibrated and the next calibration due date, or a control system is used that identifies to potential users any equipment beyond the calibration due date.

1.7.12.2.4

Measuring and test equipment is calibrated at specified intervals. These intervals are based on the frequency of use, stability characteristics and other conditions that could adversely affect the required measurement accuracy. Calibration standards are traceable to nationally recognized standards; or where such standards do not exist, provisions are established to document the basis for calibration.

The primary standards used to calibrate secondary standards have, except in certain instances, an accuracy of at least four (4) times the required accuracy of the secondary standard. In those cases where the four (4) times accuracy cannot be achieved, the basis for acceptance is documented and is authorized by the responsible manager. The secondary standards have an accuracy that assures equipment being calibrated will be within required tolerances. The basis for acceptance is documented and authorized by the responsible manager.

1.7.12.2.5

Cook Nuclear Plant procedures define the requirements for the control of standards, test equipment and process equipment.

1.7.12.2.6

When measuring and testing equipment used for inspection and testing is found to be outside of required accuracy limits at the time of calibration, evaluations are conducted to determine the validity of the

results obtained since the most recent calibration. Retests, or reinspections, are performed on suspect items. The results of evaluations are documented.

1.7.13 HANDLING, STORAGE, AND SHIPPING

1.7.13.1 SCOPE

Activities with the potential for causing contamination or deterioration, by environmental conditions such as temperature or humidity that could adversely affect the ability of an item to perform its safety-related functions and activities necessary to prevent damage or loss, are identified and controlled. These activities are cleaning, packaging, preserving, handling, shipping and storing. Controls are effected through the use of appropriate procedures and instructions.

1.7.13.2 IMPLEMENTATION

1.7.13.2.1

Procedures are used to control the cleaning, handling, storing, packaging, preserving and shipping of materials, components and systems in accordance with designated procurement requirements. These procedures include, but are not limited to, the following functions:

a) Cleaning - to assure that required cleanliness levels are achieved and maintained.

b) Packaging and preservation - to provide adequate protection against damage or deterioration. When necessary, these procedures provide for special environments, such as inert gas atmosphere, specific moisture content levels and temperature levels.

c) Handling - to preclude damage or safety hazards.

d) Storing - to minimize the possibility of loss, damage or deterioration of items in storage, including consumables such as chemicals, reagents and lubricants.

1.7.13.2.2

Controls have been established for limited shelf life items such as "O" rings, epoxy, lubricants, solvents and chemicals to assure they are correctly identified, stored and controlled to prevent shelf life expired materials from being used in the Cook Nuclear Plant. Controls are established in plant procedures.

1.7.13.2.3

Packaging and shipping requirements are provided to vendors in AEPSC Specifications (DCCs) which are a part of the procurement document, or are otherwise specified in the procurement document. Controls for receipt inspection, damaged items and special handling requirements at the Cook Nuclear Plant are established by plant procedures. Special controls are provided to assure that stainless steel components and materials are handled with approved lifting slings.

1.7.13.2.4

Storage and surveillance requirements have been established to assure segregation of storage. Special controls have been implemented for critical, high value, or perishable items. Routine surveillance is conducted on stored material to provide inspection for damage, rotation of stored pumps and motors, inspection for protection of exposed surfaces and cleanliness of the storage area.

1.7.13.2.5

Special handling procedures have been implemented for the processing of nuclear fuel during refueling outages. These procedures minimize the

risk of damage to the new and spent fuel and the possible release of radioactive material when placing the spent fuel into the spent fuel pool.

1.7.14 INSPECTION, TEST, AND OPERATING STATUS

1.7.14.1 SCOPE

Operating status of structures, systems and components is indicated by tagging of valves and switches, or by other specified means, in such a manner as to prevent inadvertent operation. The status of inspections and tests performed on individual items is clearly indicated by markings and/or logging under strict procedural controls to prevent inadvertent bypassing of such inspections and tests.

1.7.14.2 IMPLEMENTATION

1.7.14.2.1

For design change activities, including item fabrication, installation and test, a program exists which specifies the degree of control required for the identification of inspection and test status of structures, systems and components.

Physical identification is used to the extent practical to indicate the status of items requiring inspections, tests, or examinations. Procedures exist which provide for the use of calibration and rejection stickers, tags, stamps and other forms of identification to indicate test and inspection status. The Clearance Permit System uses various tags to identify equipment and system operability status. Another program establishes a tagging system for lifted leads, etc. For those items requiring calibration, the program provides for physical indication of calibration status by calibration stickers, or a control system is used.

1.7.14.2.2

Application and removal of inspection and welding stamps, and of such status indicators as tags, markings, labels, etc., is controlled by plant procedures.

The inspection status of materials received at the Cook Nuclear Plant is identified in accordance with established instructions. The status is identified as Hold, Hold for Quality Control Clearance, Reject, or Accept.

The inspection status of work in progress is controlled by the use of hold points in procedures. Plant Quality Control, or departmental ANSI N18.1 qualified personnel (reference 1.7.10.2.4 herein), inspect an activity at various stages and sign off the procedural inspection steps.

The status of welding is controlled through the use of a weld data block which identifies the inspection and NDE status of each weld.

1.7.14.2.3

Required surveillance test procedures are defined in PMIs. These instructions provide for documenting bypassed tests and rescheduling of the test.

The status of testing after minor maintenance is recorded as part of the Job Order. The status of testing after major maintenance is included as part of the procedure, and includes the performance of functional testing and approval of data by supervisory personnel.

Testing, inspection and other operations important to safety are conducted in accordance with properly reviewed and approved procedures. The PMI for plant procedures requires that procedures be followed as written. Alteration to the sequence of a procedure can only be accomplished by a procedure change which is subject to the same controls

as the original review and approval. When an immediate procedure change is required to continue in-process work or testing and the required complete review and approval process cannot be accomplished, an "On The Spot" change is processed in accordance with the PMI on plant procedures.

1.7.14.2.4

Nonconforming, inoperable, or malfunctioning structures, systems and components are clearly identified by tags, stickers, stamps, etc., and documented to prevent inadvertent use.

1.7.15 NONCONFORMING ITEMS

1.7.15.1 SCOPE

Materials, parts, or components that do not conform to requirements are controlled in order to prevent their inadvertent use. Nonconforming items are identified, documented, segregated when practical and dispositioned. Affected organizations are notified of nonconformances.

1.7.15.2 IMPLEMENTATION

1.7.15.2.1

Items, services, or activities that are deficient in characteristic, documentation, or procedure, which render the quality unacceptable or indeterminate, are identified as nonconforming and any further use is controlled. Nonconformances are documented and dispositioned, and notification is made to affected organizations. Personnel authorized to disposition, conditionally release and close out nonconformances are designated.

The Job Order System and/or the Condition/Problem Reports (refer to 1.7.16 herein) are used at Cook Nuclear Plant to identify nonconforming items and initiate corrective action for items which are installed or have been released to the Cook Nuclear Plant. Systems, components, or

materials which require repair or inspection are controlled under the Job Order System. In addition, the various procedures identified in 1.7.14 herein provide for identification, segregation and documentation of nonconforming items.

1.7.15.2.2

Nonconforming items are identified by marking, tagging, segregating, or by documented administrative controls. Documentation describes the nonconformance, the disposition of the nonconformance and the inspection requirements. It also includes signature approval of the disposition.

Completed Job Orders are reviewed by the supervisor responsible for accomplishing the work, and the supervisor of the department/section that originated the Job Order. The QA Division periodically audits the Job Order System, and on a sample basis, Job Orders.

1.7.15.2.3

Items that have been repaired or reworked are inspected and tested in accordance with the original inspection and test requirements, or alternatives, that have been documented.

Items that have the disposition of "repair" or "use-as-is" require documentation justifying acceptability. The changes are recorded to denote the as-built condition.

When required by established procedures, surveillance or operability tests are conducted on an item after rework, repair or replacement.

1.7.15.2.4

Disposition of conditional released items are closed out before the items are relied upon to perform safety-related functions.

1.7.16 CORRECTIVE ACTION

1.7.16.1 SCOPE

Conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are identified promptly and corrected as soon as practical.

For significant conditions adverse to quality, the cause of the condition is determined, corrective action is taken to correct the immediate problem, and preventive action is implemented to prevent recurrence. In these cases, the condition, cause and corrective action taken is documented and reported to appropriate levels of management.

1.7.16.2 IMPLEMENTATION

1.7.16.2.1

Procedures are established that describe the plant and AEPSC corrective action programs. These procedures are reviewed and concurred with by the AEPSC QA Division.

1.7.16.2.2

Condition/Problem Reports provide the mechanism for plant and AEPSC personnel to notify management of conditions adverse to quality. Condition/Problem Reports are also used to report violations to codes, regulations and the Technical Specifications. Investigations of reported conditions adverse to quality are assigned by management. The Condition/Problem Report is used to document the investigation of a problem; and to identify the need for a design change to correct system or equipment deficiencies, or to identify the need for the initiation of Job Orders to correct minor deficiencies. Further, Condition/Problem Reports are used to identify those actions necessary to prevent recurrence of the reported condition.

Significant problems, which are so designated on Con..tion/Problem Reports, are reviewed by the PNSRC for evaluation of actions taken, or being taken, to correct the deficiency and prevent recurrence.

The AEPSC NSDRC is responsible for assuring that independent reviews of violations (as specified in the Technical Specifications) are performed. These violations are considered significant problems which are documented on Condition/Problem Reports. The reviews will provide an independent evaluation of the reported problems and corrective actions.

The AEPSC QA Division periodically audits the corrective action systems for compliance and effectiveness.

1.7.17 QUALITY ASSURANCE RECORDS

1.7.17.1 SCOPE

Records that furnish evidence of activities affecting the quality of safety-related structures, systems and components are maintained. They are accurate, complete, legible and are protected against damage, deterioration, or loss. They are identifiable and retrievable.

1.7.17.2 IMPLEMENTATION

1.7.17.2.1

Documents that furnish evidence of activities affecting the quality of safety-related items are generated and controlled in accordance with the procedure that governs those activities. Upon completion, these documents are considered records. These records include:

a) Results of reviews, inspections, surveillances, tests, audits and material analyses.
b) Qualification of personnel, procedures and equipment.
c) Operation logs.
d) Maintenance and modification procedures and related inspection results.

e) Reportable occurrences.

f) Records required by the plant Technical Specifications.

g) Problem Reports.

h) Other documentation such as drawings, specifications, dedication plans, procurement documents, calibration procedures and reports.

i) Radiographs.

1.7.17.2.2

Instructions and procedures establish the requirements for the identification and preparation of records for systems and equipment under the QA Program, and provide the controls for retention of these records.

Criteria for the storage location of quality related records, and a retention schedule for these records, has been established.

File Indexes have been established to provide direction for filing, and to provide for the retrievability of the records.

Controls have been established for limiting access to the Plant Master File to prevent unauthorized entry, unauthorized removal, and for use of the records under emergency conditions. The Records Management Supervisor is responsible for the control and operation of the Plant Master File Room.

1.7.17.2.3

Within AEPSC, each department/division manager is responsible for the identification, collection, maintenance and storage of records generated by their department/division. Procedures ensure the maintenance of records sufficient to furnish objective evidence that activities affecting quality are in compliance with the established QA Program.

1.7.17.2.4

When a document becomes a record, it is designated as permanent, or nonpermanent, and then transmitted to file. Nonpermanent records have specified retention times. Permanent records are maintained for the life of the plant or equipment, as applicable.

1.7.17.2.5

Only authorized personnel may issue corrections or supplements to records.

1.7.17.2.6

Traceability between the record and the item or activity to which it applies is provided.

1.7.17.2.7

Except for records that can only be stored as originals, such as radiographs and some strip charts, or micrographs thereof, records are stored in remote, dual facilities to prevent damage, deterioration, or loss due to natural or unnatural causes. When only the single original can be retained, special fire-rated facilities are used.

1.7.18 AUDITS

1.7.18.1 SCOPE

A comprehensive system of audits is carried out to provide independent evaluation of compliance with, and the effectiveness of, the QA Program, including those elements of the program implemented by suppliers and contractors. Audits are performed in accordance with written procedures or checklists by qualified personnel not having direct responsibility in the areas audited. Audit results are documented and reviewed by management. Follow-up action is taken where indicated.

1.7.18.2 IMPLEMENTATION

1.7.18.2.1 AEPSC QA Division Responsibilities

The basic responsibility for the assessment of the QA Program is vested in the AEPSC QAD. The AEPSC QAD is primarily responsible for ensuring that proper QA programs are established and for verification of their implementation. These responsibilities are discharged in cooperation with the AEPSC and Cook Nuclear Plant management and their staffs.

1.7.18.2.2

Internal audits are performed in accordance with established schedules that reflect the status and importance of safety to the activities being performed. All areas where the requirements of 10CFR50, Appendix B apply are audited within a period of one to two years.

1.7.18.2.3

The AEPSC QAD conducts audits to verify the adequacy and implementation of the QA Program at the Cook Nuclear Plant and within the AEP System. QA audit reports are distributed to the appropriate management and the NSDRC (all audits).

1.7.18.2.4

The independent off-site review and audit organization is the AEPSC NSDRC. This committee is composed of AEPSC, I&M and Cook Nuclear Plant management members. An NSDRC Manual has been developed for this committee which contains the NSDRC Charter and procedures. The NSDRC conducts periodic audits of Cook Nuclear Plant operations pursuant to established criteria (Technical Specifications, etc.).

NSDRC audit reports are submitted for review to the NSDRC membership, the Chairman of the NSDRC, and the AEPSC Senior Executive Vice President

- Engineering and Construction. Problem Reports provide for the recording of actions taken to correct deficiencies found during these audits.

1.7.18.2.5

The Cook Nuclear Plant on-site review group is the PNSRC. This committee reviews plant operations as a routine evaluation and serves to advise the Plant Manager on matters related to nuclear safety. The composition of the committee is defined in the Technical Specifications.

The PNSRC also reviews instructions, procedures, and design changes for safety-related systems prior to approval by the Plant Manager. In addition, this committee serves to conduct investigations of violations to Technical Specifications, and reviews significant Problem Reports to determine if appropriate action has been taken.

1.7.18.2.6

Audits of suppliers and contractors are scheduled based on the status of safety importance of the activities being performed, and are initiated early enough to assure effective quality assurance during design, procurement, manufacturing, construction, installation, inspection and testing.

Principal contractors are required to audit their suppliers systematically in accordance with the criteria established within their quality assurance programs.

1.7.18.2.7

Regularly scheduled audits are supplemented by "special audits" when significant changes are made in the QA Program, when it is suspected that quality is in jeopardy, or when an independent assessment of program effectiveness is considered necessary.

1.7.18.2.8

Audits include an objective evaluation of practices, procedures, instructions, activities and items related to quality; and a review of documents and records to confirm that the QA Program is effective and properly implemented.

1.7.18.2.9

Audit procedures and the scope, plans, checklists and results of individual audits are documented.

1.7.18.2.10

Personnel selected for auditing assignments have experience, or are given training commensurate with the needs of the audit, and have no direct responsibilities in the areas audited.

1.7.18.2.11

Management of the audited organization identifies and takes appropriate action to correct observed deficiencies and to prevent recurrence. Follow-up is performed by the auditing organization to ensure that the appropriate actions were taken. Such follow-up includes reaudits, when necessary.

1.7.18.2.12

The adequacy of the QA Program is regularly assessed by AEPSC management. The following activities constitute formal elements of that assessment:

a) Audit reports, including follow-up on corrective action accomplishment and effectiveness, are distributed to appropriate levels of management.

b) Individuals independent from the QA organization, but knowledgeable in auditing and quality assurance, periodically review the effectiveness of the QA Programs. Conclusions and recommendations are reported to the AEPSC Vice President - Nuclear Operations.

1.7.19 FIRE PROTECTION QA PROGRAM

1.7.19.1 Introduction

The Cook Nuclear Plant Fire Protection QA Program has been developed using the guidance of NRC Branch Technical Position (APCSB) 9.5-1, Appendix A, Section C, "Quality Assurance Program," and NRC clarification "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance," dated June 14, 1977. As such, the Fire Protection QA Program is part of the overall QA Program for the plant. The Fire Protection QA Program encompasses design, procurement, fabrication, construction, surveillance, inspection, operation, maintenance, modification, and audits.

Implementation and assessment of the Fire Protection QA Program is the responsibility of each involved AEPSC and Indiana Michigan Power Company organization.

1.7.19.2 Organization

The Fire Protection QA Program is under the management control of AEPSC. This control consists of:

1) Verifying the effectiveness of the Fire Protection QA Program through review, surveillance, and audits.

2) Directing formulation, implementation, and assessment of the Fire Protection QA Program by procedural controls.

3) Assuring the QA program is acceptable to the management responsible for fire protection.

The Plant Manager has delegated responsibility to various Cook Nuclear Plant departments for the following fire protection activities:

a) Maintenance of fire protection systems.
b) Testing of fire protection equipment.
c) Fire safety inspections.
d) Fire fighting procedures.
e) Fire drills.
f) Emergency remote shutdown procedures.
g) Emergency repair procedures (10CFR50, Appendix R).

The Fire Protection QA Program at the Cook Nuclear Plant also provides for inspection of fire hazards, explosion hazards, and training of fire brigade and responding fire departments.

The Safety and Assessment Department's Fire Protection Shift Supervisor on duty, or designee, is designated as the Fire Brigade Leader and coordinates the fire fighting efforts of shift personnel and the Fire Brigade. The Operations Department provides an individual with plant systems knowledge to serve as an advisor to the Fire Brigade Leader.

1.7.19.3 Design Control and Procurement Document Control

Quality standards are specified in the design documents such as appropriate fire protection codes and standards, and, as necessary, deviations and changes from these quality standards are controlled.

The Cook Nuclear Plant design was reviewed by qualified personnel to ensure inclusion of appropriate fire protection requirements. These reviews include items such as:

1) Verification as to the adequacy of electrical isolation and cable separation criteria.

2) Verification of appropriate requirements for room isolation (sealing penetrations, floors and other fire barriers).

3) Determination for increase in fire loadings.

4) Determination for the need of additional fire detection and suppression equipment.

Procurement of fire protection equipment and related items are subject to the reauirements of the fire protection procurement documents. A review of these documents is performed to assure fire protection requirements and quality requirements are correctly stated, verifiable, and controllable, and that there is adequate acceptance and rejection criteria. Procurement documents must be prepared, reviewed, and approved according to QA Program requirements.

Design and procurement document changes, including field changes and design deviations, are controlled by procedure.

1.7.19.4 Instructions, Procedures and Drawings

Inspections, tests, administrative controls, fire drills and training that assist in implementing the fire protection program are prescribed by approved instructions or procedures.

Indoctrination and training programs for fire prevention and fire fighting are implemented in accordance with approved procedures. Activities associated with the fire protection systems and fire

protection related systems are prescribed and accomplished in accordance with documented instructions, procedures, and drawings. Instructions and procedures for design, installation, inspection, tests, maintenance, modification and administrative controls are reviewed through audits to assure that the fire protection program is maintained.

Operation and maintenance information has been provided to the plant in the form of System Descriptions and equipment supplier information.

1.7.19.5 Control of Purchased Items and Services

Measures are established to assure that purchased items and services conform to procurement documents. These measures include provisions, as appropriate, for source evaluation and selection, objective evidence of quality furnished by the contractor, inspections at suppliers, or receipt inspection.

Source or receipt inspection is provided, as a minimum, for those items where quality cannot be verified after installation.

1.7.19.6 Inspection

A program for independent inspection of the fire protection activities has been established and implemented.

These inspections are performed by personnel other than those responsible for implementation of the activity. The inspections include:

a) Inspection of installation, maintenance and modification of fire protection systems and equipment.

b) Inspections of penetration seals and fire retardant coating installations to verify the activity is satisfactorily completed in accordance with installation specifications.

c) Inspections of cable routing to verify conformance with design requirements as specified in AEPSC Specifications and/or plant procedures.

d) Inspections to verify that appropriate requirements for fire barriers are satisfied following installation, modification, repair or replacement activities.

e) Measures to assure that inspection personnel are independent from the individuals performing the activity being inspected and are knowledgeable in the design and installation requirements for fire protection.

f) Inspection procedures, instructions or checklists for required inspections.

g) Periodic inspections of fire protection systems, emergency breathing and auxiliary equipment.

h) Periodic inspections of materials subject to degradation, such as fire stops, seals and fire retardant coating as required by Technical Specifications or manufacturer's recommendations.

1.7.19.7 <u>Test and Test Control</u>

a) Installation testing - Following installation, modification, repair, or replacement, sufficient testing is performed to demonstrate that the fire protection systems and equipment will perform satisfactorily. Written test procedures for installation tests incorporate the requirements and acceptance limits contained in applicable design documents.

b) Periodic testing - Periodic testing occurs to document that fire protection equipment functions in accordance with its design.

c) Programs have been established to verify the testing of fire protection systems, and to verify that test personnel are effectively trained.

d) Test results are documented, evaluated, and their acceptability determined by a qualified responsible individual or group.

1.7.19.8 Inspection, Test and Operating Status

The inspection, test and operating status for plant Technical Specification fire protection systems are performed as described in 1.7.14 herein.

1.7.19.9 Nonconforming Items

Technical Specification fire protection equipment nonconformances are identified and dispositioned as described in 1.7.15 herein.

1.7.19.10 Corrective Action

The corrective action mechanism described in 1.7.16 herein applies to the Technical Specification fire protection equipment.

1.7.19.11 Records

Records generated to support the fire protection program are controlled as described in 1.7.17 herein.

1.7.19.12 Audits

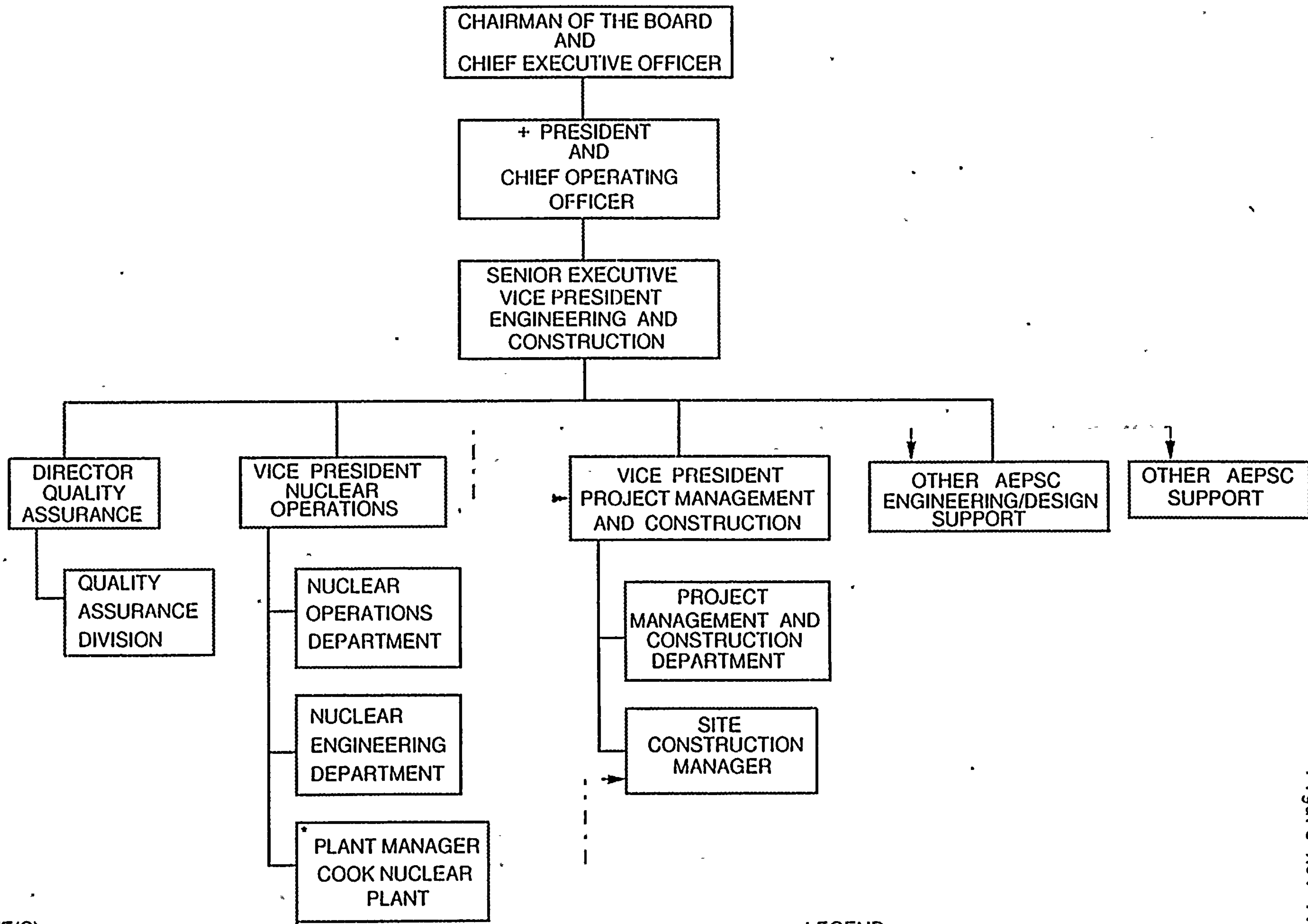
Audits are conducted and documented to verify compliance with the Fire Protection QA Program as described in 1.7.1.18 herein.

Audits are periodically performed to verify compliance with the administrative controls and implementation of fire protection quality

assurance criteria. The audits are performed in accordance with pre-established written procedures or checklists. Audit results are documented and reviewed by management having responsibility in the area audited. Follow-up action is taken by responsible management to correct the deficiencies revealed by the audit.

AMERICAN ELECTRIC POWER SERVICE CORPORATION

Support Organization for the Cook Plant



1.7-88

NOTE(S)

* NOT PART OF AEPSC ORGANIZATION - SHOWN FOR INFORMATION ONLY

+ EFFECTIVE MA[illegible] 1, 1992

LEGEND

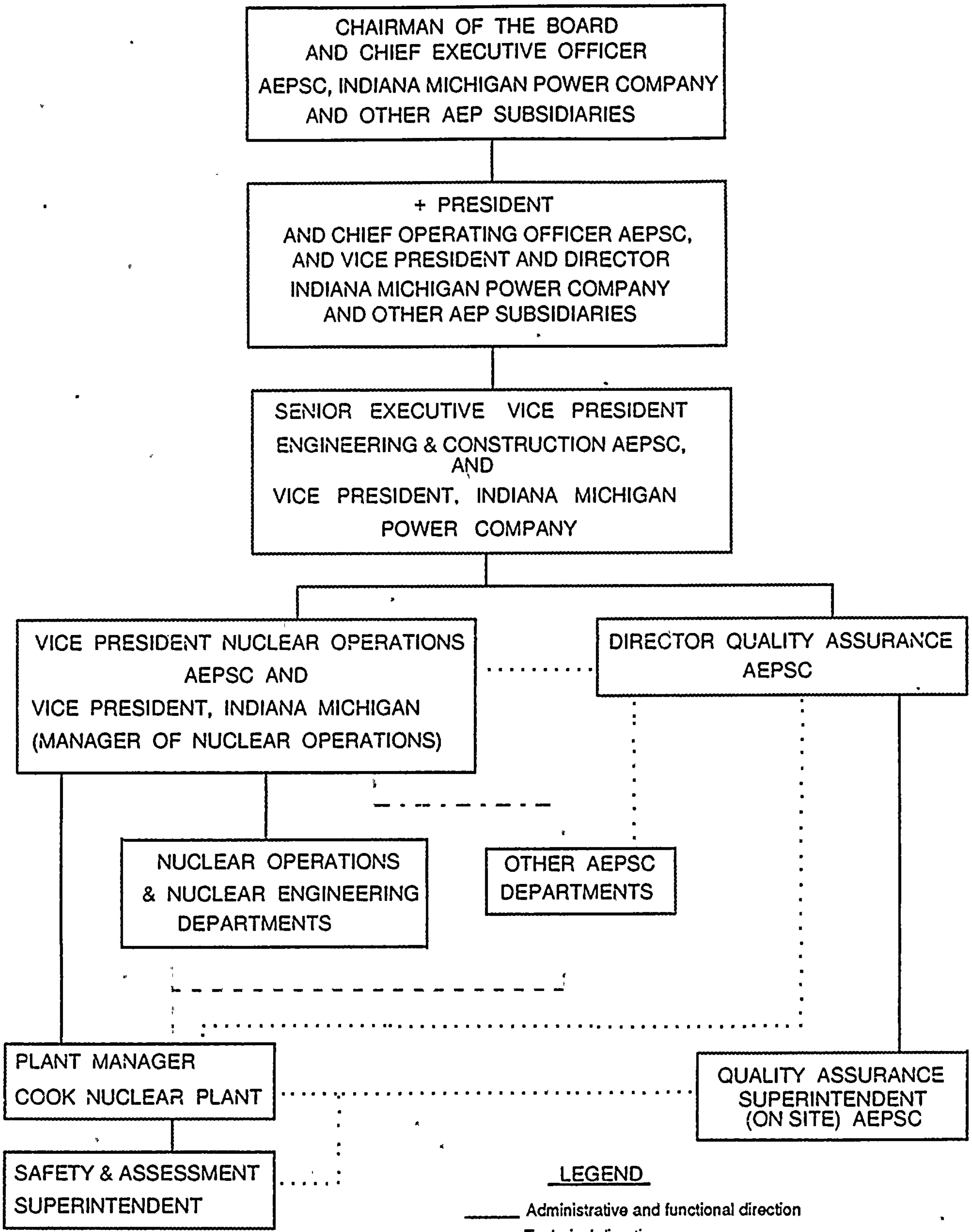
ADMINISTRATIVE AND FUNCTIONAL DIRECTION

FUNCTIONAL DIRECTION FOR THE COOK NUCLEAR PLANT

Figure No. 1.7-1

July, 1992

ORGANIZATIONAL RELATIONSHIPS WITHIN THE AMERICAN ELECTRIC POWER SYSTEM PERTAINING TO QUALITY ASSURANCE AND QUALITY CONTROL SUPPORT OF THE COOK NUCLEAR PLANT



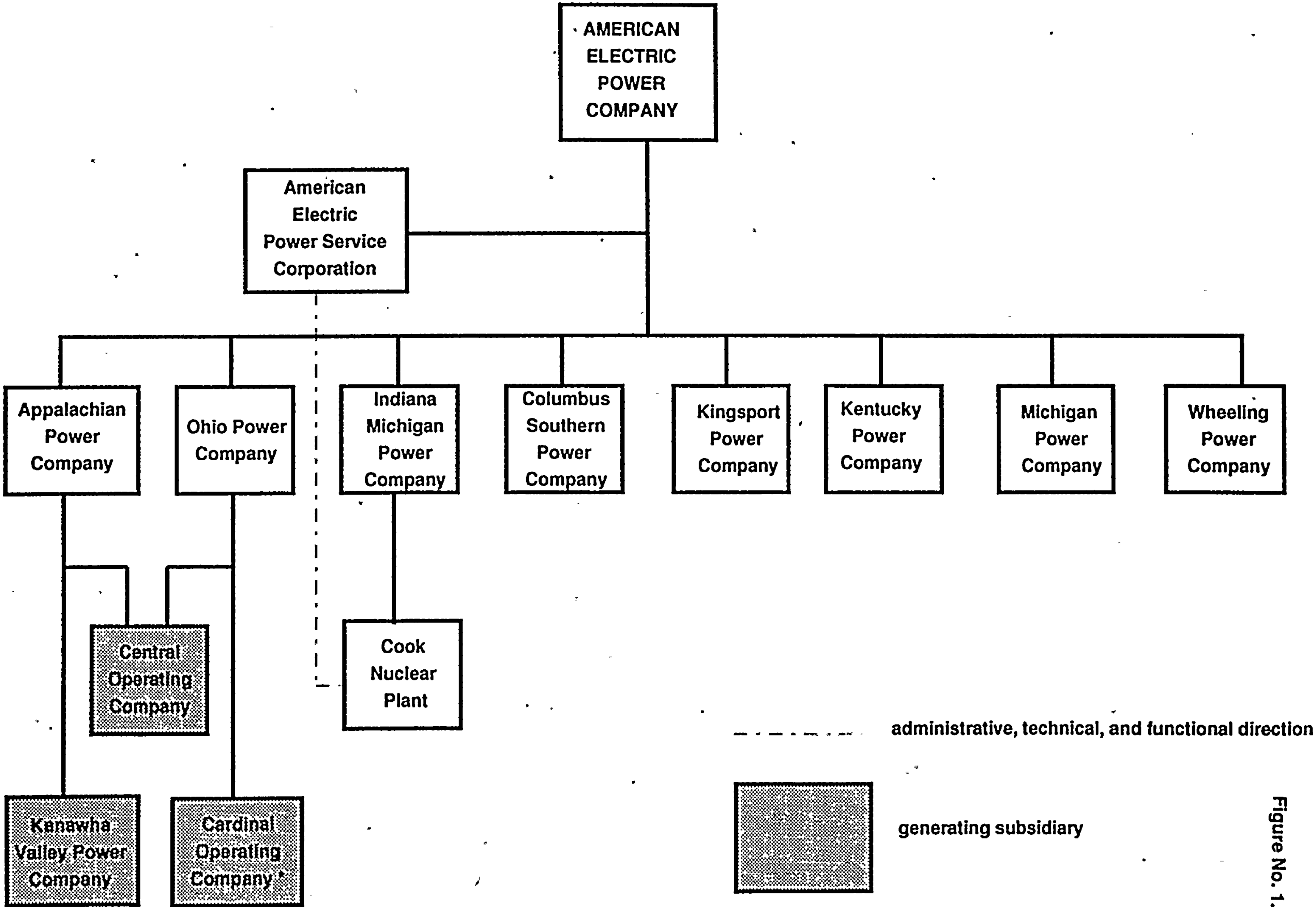
LEGEND

- ______ Administrative and functional direction
- _ _ _ Technical direction
- . . . . . . Technical liaison
- _ . _ Functional direction for Cook Nuclear Plant Activities

NOTES

+ EFFECTIVE MARCH 1, 1992

American Electric Power Company



* jointly owned with Buckeye Power, Inc.

AEPSC QUALITY ASSURANCE DIVISION ORGANIZATION

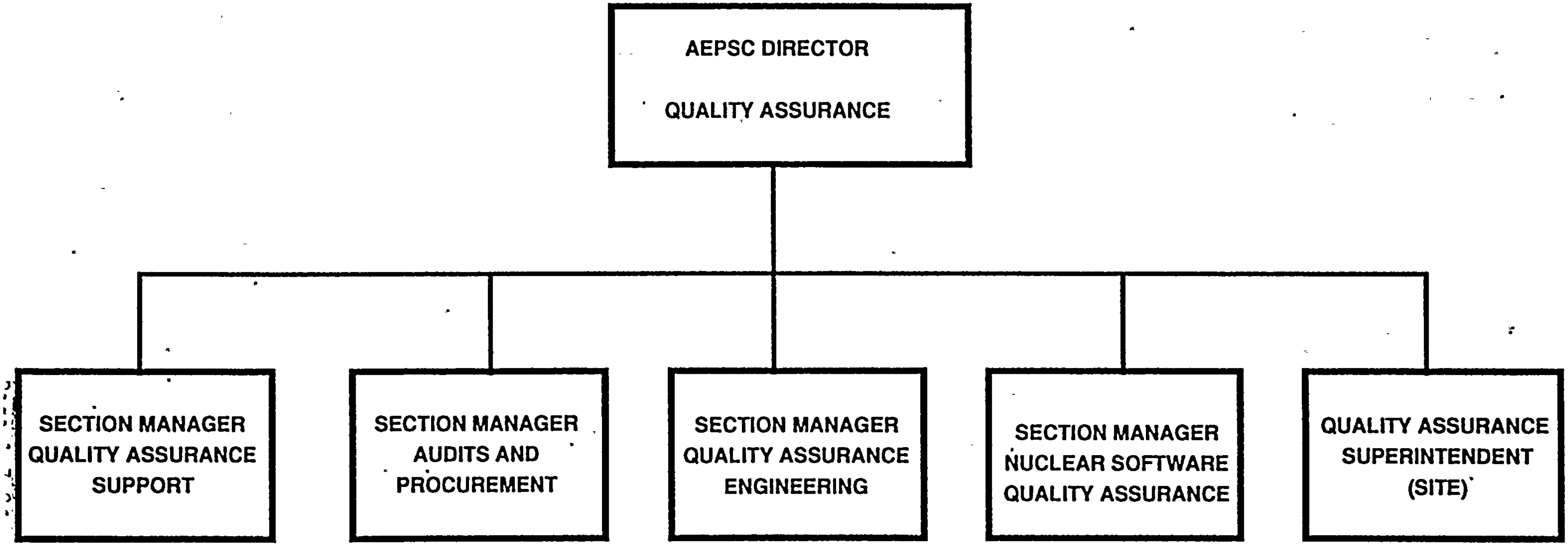


Figure No. 1.7-4

Organization for the Cook Nuclear Plant
Indiana Michigan Power Company

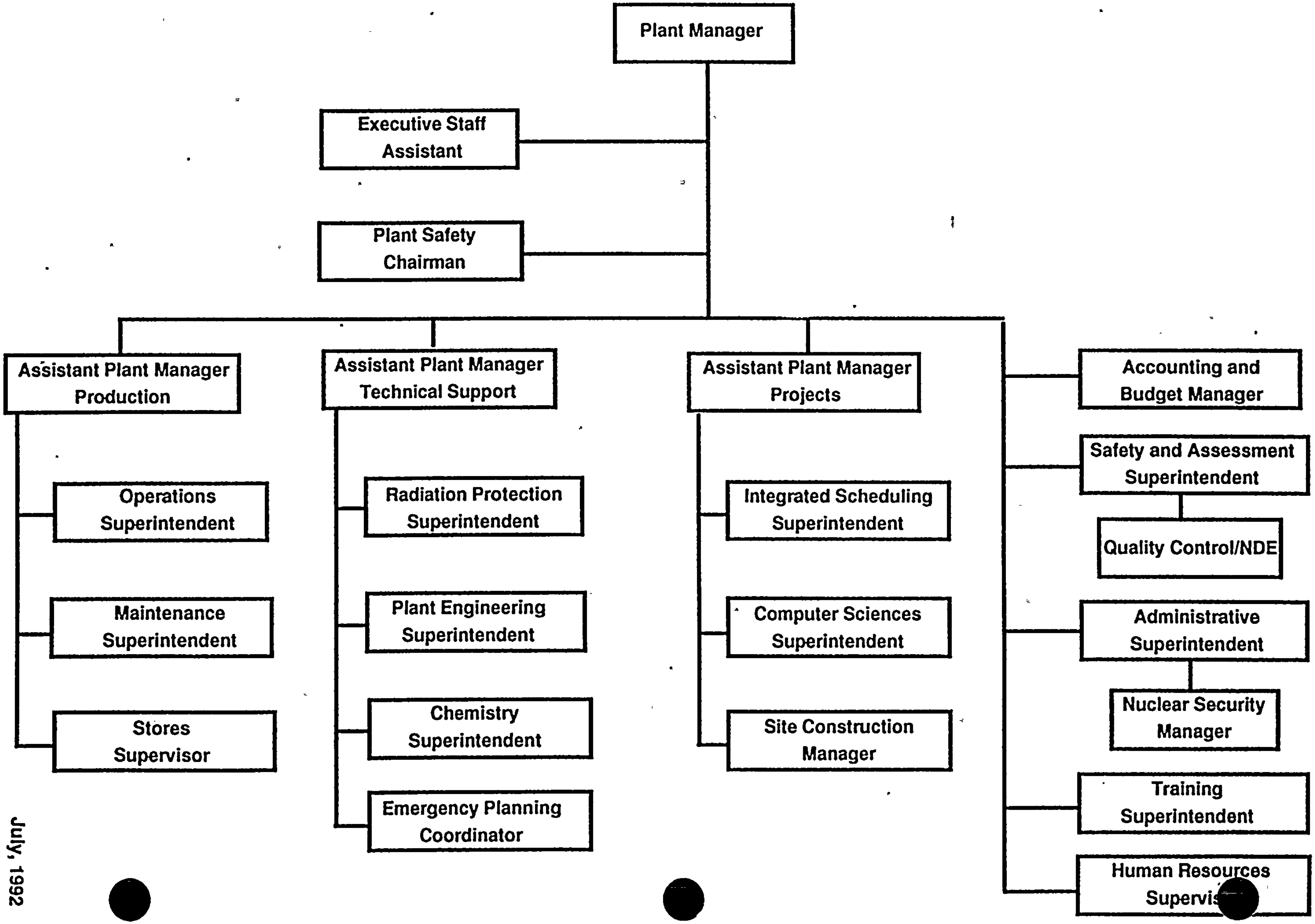


Figure No. 1.7-5

APPENDIX A

REGULATORY AND SAFETY GUIDES/ANSI STANDARDS

1.	Reg. Guide 1.8 (9/75)	- Personnel Selection and Training
	ANSI N18.1 (1971)	- Selection and Training of Nuclear Power Plant Personnel
2.	Reg. Guide 1.14 (8/75)	- Reactor Coolant Pump Flywheel Integrity
3.	Reg. Guide 1.16 (8/75)	- Reporting of Operating Information, Appendix A - Technical Specifications
4.	Safety Guide 30 (8/72)	- Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment
	ANSI N45.2.4 (1972)	- Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations
5.	Reg. Guide 1.33 (02/78)	- Quality Assurance Program Requirements (Operation)
	ANSI N18.7 (1976) (ANS 3.2 1976)	- Administrative Controls and Quality Assurance for the Operational Phase of Nuclear Power Plants
	ANSI N45.2 (1977)	- Quality Assurance Program Requirements for Nuclear Facilities

6.	Reg. Guide 1.37 (3/73)	-	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants
	ANSI N45.2.1 (1973)	-	Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants
7.	Reg. Guide 1.3 (10/76)	-	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants
	ANSI N45.2.2 (1972)	-	Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants (During the Construction Phase)
8.	Reg. Guide 1.39 (10/76)	-	Housekeeping Requirements for Water-Cooled Nuclear Power Plants
	ANSI N45.2.3 (1973)	-	Housekeeping During the Construction Phase of Nuclear Power Plants
9.	Reg. Guide 1.54 (6/73)	-	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants
	ANSI N101.4 (1972)	-	Quality Assurance for Protective Coatings Applied to Nuclear Facilities
10.	Reg. Guide 1.58 (9/80)	-	Qualification of Nuclear Power Plant Inspection, Examination and Testing Personnel

ANSI N45.2.6 (1978)	-	Qualifications of Inspection, Examination, and Testing Personnel for Nuclear Power Plants
11. Reg. Guide 1.63 (7/78)	-	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants
12. Reg. Guide 1.64 (10/73)	-	Quality Assurance Requirements for the Design of Nuclear Power Plants
ANSI N45.2.11 (1974)	-	Quality Assurance Requirements for the Design of Nuclear Power Plants
13. Reg. Guide 1.74 (2/74)	-	Quality Assurance Terms and Definitions
ANSI N45.2.10 (1973)	-	Quality Assurance Terms and Definitions
14. Reg. Guide 1.88 (10/76)	-	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
ANSI N45.2.9 (1974)	-	Requirements for Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants
15. Reg. Guide 1.94 (4/76)	-	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

	ANSI N45.2.5 (1974)	- Supplementary Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
16.	Reg. Guide 1.108 (8/77)	- Periodic Testing of Diesel Generator Units used as Onsite Electric Power Systems at Nuclear Power Plants
17.	Reg. Guide 1.123 (7/77)	- Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
	ANSI N45.2.13 (1976)	- Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants
18.	Reg. Guide 1.144 (1/79)	- Auditing of Quality Assurance Programs for Nuclear Power Plants
	ANSI N45.2.12 (1977)	- Requirements for Auditing of Quality Assurance Programs for Nuclear Power Plants
19.	Reg. Guide 1.146 (8/80)	- Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants
	ANSI N45.2.23 (1978)	- Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

20. ANSI N45.2.8 (1975) - Supplementary Quality Assurance Requirements for Installation, Inspection and Testing of Mechanical Equipment and Systems for the Construction Phase of Nuclear Power Plants

21. ANSI N45.4 (1972) - Leakage-Rate Testing of Containment Structures for Nuclear Reactors

APPENDIX B

AEPSC/I&M EXCEPTIONS TO OPERATING PHASE
STANDARDS AND REGULATORY GUIDES

1. GENERAL

Requirement

Certain Regulatory Guides invoke, or imply, Regulatory Guides and standards in addition to the standard each primarily endorses.

Certain ANSI Standards invoke, or imply, additional standards.

Exception/Interpretation

The AEPSC/I&M commitment refers to the Regulatory Guides and ANSI Standards specifically identified in Appendix A. Additional Regulatory Guides, ANSI Standards and similar documents implied, or referenced, in those specifically identified are not part of this commitment.

2. N18.7, General

Exception/Interpretation

AEPSC and I&M have established both an on-site and off-site standing committee for independent review activities; together they form the independent review body.

The standard numeric and qualification requirement may not be met by each group individually. Procedures will be established to specify how each group will be involved in review activities. This exception/interpretation is consistent with the plant's Technical Specifications.

2a. Sec. 4.3.1

Requirement

"Personnel assigned responsibility for independent reviews shall be specified in both number and technical disciplines, and shall collectively have the experience and competence required to review problems in the following areas: ...."

Exception/Interpretation

AEPSC Nuclear Safety and Design Review Committee (NSDRC) and Plant Nuclear Safety Review Committee (PNSRC) will not have members specified by number, nor by technical disciplines, and its members may not have the experience and competence required to review problems in all areas listed in this section. This exception/interpretation is consistent with the plant's Technical Specifications.

The NSDRC and PNSRC will not specifically include a member qualified in nondestructive testing, but will use qualified technical consultants to perform this and other functions as determined necessary by the respective committee chairman.

2b. Sec. 4.3.2.1

Requirement

"When a standing committee is responsible for the independent review program, it shall be composed of no less than five persons of whom no more than a minority are members of the on-site operating organization. Competent alternates are permitted if designated in advance. The use of alternates shall be restricted to legitimate absences of principals."

Exception/Interpretation

See Item 2a.

2c. Sec. 4.3.3.1

<u>Requirement</u>

"... recommendations ... shall be disseminated promptly to appropriate members of management having responsibility in the area reviewed."

<u>Exception/Interpretation</u>

Recommendations made as a result of review will generally be conveyed to the on-site, or off-site, standing committee. Procedures will be maintained specifying how recommendations are to be considered.

2d. Sec. 4.3.4

<u>Requirement</u>

"The following subjects shall be reviewed by the independent review body: ...."

<u>Exception/Interpretation</u>

Subjects requiring review will be as specified in the plant Technical Specifications.

2e. Sec. 4.3.4(3)

<u>Requirement</u>

"Changes in the Technical Specifications or License Amendments relating to nuclear safety are to be reviewed by the independent review body prior to implementation, except in those cases where the change is identical to a previously reviewed proposed change."

<u>Exception/Interpretation</u>

Although the usual practice is to meet this requirement, exceptions are made to NSDRC review and approval prior to implementation in rare cases with the permission of the NSDRC Chairman and Secretary. PNSRC review and approval is always done prior to implementation of Technical Specification changes.

2f. Sec. 4.4

Requirement

"The on-site operating organization shall provide, as part of the normal duties of plant supervisory personnel ...."

Exception/Interpretation

Some of the responsibilities of the on-site operating organization described in Section 4.4 may be carried out by the PNSRC and/or NSDRC as described in plant Technical Specifications.

2g. Sec. 5.2.2

Requirement

"Temporary changes, which clearly do not change the intent of the approved procedure, shall as a minimum be approved by two members of the plant staff knowledgeable in the areas affected by the procedures. At least one of these individuals shall be the supervisor in charge of the shift and hold a senior operator's license on the unit affected."

Exception/Interpretation

I&M considers that this requirement applies only to procedures identified in plant Technical Specifications. Temporary changes to these procedures shall be approved as described in plant Technical Specifications.

2h. Sec. 5.2.6

Requirement

"In cases where required documentary evidence is not available, the associated equipment or materials must be considered nonconforming in accordance with Section 5.2.14. Until suitable documentary evidence is available to show the equipment or material is in conformance, affected systems shall be considered to be inoperable and reliance shall not be placed on such systems to fulfill their intended safety functions."

Exception/Interpretation

I&M initiates appropriate corrective action when it is discovered that documentary evidence does not exist for a test or inspection which is a

requirement to verify equipment acceptability. This action includes a technical evaluation of the equipment's operability status.

2i. Sec. 5.2.

Requirement

"A surveillance testing and inspection program ... shall include the establishment of a master surveillance schedule reflecting the status of all planned in-plant surveillance tests and inspections."

Exception/Interpretation

Separate master schedules may exist for different programs, such as ISI, pump and valve testing, and Technical Specification surveillance testing.

2j. Sec. 5.2.13.1

Requirement

"To the extent necessary, procurement documents shall require suppliers to provide a Quality Assurance Program consistent with the pertinent requirements of ANSI N45.2 - 1977."

Exception/Interpretation

To the extent necessary, procurement documents require that the supplier has a documented Quality Assurance Program consistent with the pertinent requirements of 10CFR50, Appendix B; ANSI N45.2; or other nationally recognized codes and standards.

2k. Sec. 5.2.13.2

Requirement

ANSI N18.7 and N45.2.13 specify that where required by code, regulation, or contract, documentary evidence that items conform to procurement requirements shall be available at the nuclear power plant site prior to installation or use of such items.

Exception/Interpretation

The required documentary evidence is available at the site prior to use, but not necessarily prior to installation. This allows installation to

proceed while any missing documents are being obtained, but precludes dependence on the item for safety purposes.

2l. Sec. 5.2.15

<u>Requirement</u>

"Plant procedures shall be reviewed by an individual knowledgeable in the area affected by the procedure no less frequently than every two years to determine if changes are necessary or desirable."

<u>Exception/Interpretation</u>

Biennial reviews are not performed in that I&M has programmatic control requirements in place that make the biennial review process redundant from a regulatory perspective. These programmatic controls were effected in an effort to ensure that plant instructions and procedures are reviewed for possible revision when pertinent source material is revised, therefore maintaining the procedures current. We believe that this approach, in addition to an annual random sampling of procedures, better addresses the intent of the biennial review process and is more acceptable from both a technical and practical perspective than a static two-year review process.

2m. Sec. 5.2.16

<u>Requirement</u>

Records shall be made, and equipment suitably marked, to indicate calibration status.

<u>Exception/Interpretation</u>

See Item 6b.

2n. Sec. 5.3.5(4)

Requirement

This section requires that where sections of documents such as vendor manuals, operating and maintenance instructions, or drawings are incorporated directly, or by reference into a maintenance procedure, they shall receive the same level of review and approval as operating procedures.

Exception/Interpretation

Such documents are reviewed by appropriately qualified personnel prior to use to ensure that, when used as instructions, they provide proper and adequate information to ensure the required quality of work. Maintenance procedures which reference these documents receive the same level of review and approval as operating procedures.

3. N45.2.1,

3a. Sec. 3

Requirement

N45.2.1 establishes criteria for classifying items into "cleanliness levels," and requires that items be so classified.

Exception/Interpretation

Instead of using the cleanliness level classification system of N45.2.1, the required cleanliness for specific items and activities is addressed on a case-by-case basis.

Cleanliness is maintained, consistent with the work being performed, so as to prevent the introduction of foreign material. As a minimum, cleanliness inspections are performed prior to closure of "nuclear" systems and equipment. Such inspections are documented.

3b. Sec. 5

Requirement

"Fitting and tack-welded joints (which will not be immediately sealed by welding) shall be wrapped with polyethylene or other nonhalogenated plastic film until the welds can be completed."

Exception/Interpretation

I&M sometimes uses other nonhalogenated material, compatible with the parent material, since plastic film is subject to damage and does not always provide adequate protection.

4. N45.2.2, General

Requirement

N45.2.2 establishes requirements and criteria for classifying safety related items into protection levels.

Exception/Interpretation

Instead of classifying safety related items into protection levels, controls over the packaging, shipping, handling and storage of such items are established on a case-by-case basis with due regard for the item's complexity, use and sensitivity to damage. Prior to installation or use, the items are inspected and serviced, as necessary, to assure that no damage or deterioration exists which could affect their function.

4a. Sec. 3.9 and Appendix A3.9

Requirement

"The item and the outside of containers shall be marked."
(Further criteria for marking and tagging are given in the Appendix.)

Exception/Interpretation

These requirements were originally written for items packaged and shipped to construction projects. Full compliance is not always necessary in the case of items shipped to operating plants and may, in some cases,

increase the probability of damage to the item. The requirements are implemented to the extent necessary to assure traceability and integrity of the item.

4b. Sec. 5.2.2

<u>Requirement</u>

"Receiving inspections shall be performed in an area equivalent to the level of storage."

<u>Exception/Interpretation</u>

Receiving inspection area environmental controls may be less stringent than storage environmental requirements for an item. However, such inspections are performed in a manner and in an environment which do not endanger the required quality of the item.

4c. Sec. 6.2.4

<u>Requirement</u>

"The use or storage of food, drinks and salt tablet dispensers in any storage area shall not be permitted."

<u>Exception/Interpretation</u>

Packaged food for emergency or extended overtime use may be stored in material stock rooms. The packaging assures that materials are not contaminated. Food will not be "used" in these areas.

4d. Sec. 6.3.4

<u>Requirement</u>

"All items and their containers shall be plainly marked so that they are easily identified without excessive handling or unnecessary opening of crates and boxes."

<u>Exception/Interpretation</u>

See N45.2.2, Section 3.9 (Exception 4b.).

4e. Sec. 6.4.1

Requirement

"Inspections and examinations shall be performed and documented on a periodic basis to assure that the integrity of the item and its container ... is being maintained."

Exception/Interpretation

The requirement implies tnat all inspections and examinations of items in storage are to be performed on the same schedule. Instead, the inspections and examinations are performed in accordance with material storage procedures which identify the characteristics to be inspected and include the required frequencies. These procedures are based on technical considerations which recognize that inspections and frequencies needed vary from item to item.

5. N45.2.3,

5a. Sec. 2.1

Requirement

Cleanliness requirements for housekeeping activities shall be established on the basis of five zone designations.

Exception/Interpretation

Instead of the five-level zone designation system referenced in ANSI N45.2.3, I&M bases its controls over housekeeping activities on a consideration of what is necessary and appropriate for the activity involved. The controls are effected through procedures or instructions. Factors considered in developing the procedures and instructions include cleanliness control, personnel safety, fire prevention and protection, radiation control and security. The procedures and instructions make use of standard janitorial and work practices to the extent possible. However, in preparing these procedures, consideration is also given to the recommendations of Section 2.1 of ANSI N45.2.3.

6. N45.2.4,

6a. Sec. 2.2

Requirement

Section 2.2 establishes prerequisites which must be met before the installation, inspections and testing of instrumentation and electrical equipment may proceed. These prerequisites include personnel qualification, control of design, conforming and protected materials and availability of specified documents..

Exception/Interpretation

During the operations phase, this requirement is considered to be applicable to modifications and initial start-up of electrical equipment. For routine or periodic inspection and testing, the prerequisite conditions will be achieved, as necessary.

6b. Sec. 6.2.1

Requirement

"Items requiring calibration shall be tagged or labeled on completion, indicating date of calibration and identity of person that performed calibration."

Exception/Interpretation

Frequently, physical size and/or location of installed plant instrumentation precludes attachment of calibration labels or tags. Instead, each instrument is uniquely identified and is traceable to its calibration record.

A scheduled calibration program assures that each instrument's calibration is current.

7. N45.2.5,

7a. Sec. 2.5.2

Requirement

"When discrepancies, malfunctions or inaccuracies in inspection and testing equipment are found during calibration, all items inspected with

that equipment since the last previous calibration shall be considered unacceptable until an evaluation has been made by the responsible authority and appropriate action taken."

Exception/Interpretation

I&M uses the requirements of N18.7, Section 5.2.16, rather than N45.2.5, Section 2.5.2. The N18.7 requirements are more applicable to an operating plant.

7b. Sec. 5.4

Requirement

"Hand torque wrenches used for inspection shall be controlled and must be calibrated at least weekly and more often if deemed necessary. Impact torque wrenches used for inspection must be calibrated at least twice daily."

Exception/Interpretation

Torque wrenches are controlled as measuring and test equipment in accordance with ANSI N18.7, Section 5.2.16. Calibration intervals are based on use and calibration history rather than as per N45.2.5.

8. N45.2.6, Sec. 1.2

Requirement

"The requirements of this standard apply to personnel who perform inspections, examinations and tests during fabrication prior to or during receipt of items at the construction site, during construction, during preoperational and start-up testing and during operational phases of nuclear power plants."

Exception/Interpretation

Personnel participating in testing who take data or make observations, where special training is not required to perform this function, need not be qualified in accordance with ANSI N45.2.6, but need only be trained to the extent necessary to perform the assigned function.

9. Req. Guide 1.58 - General

Requirement

Qualification of nuclear power plant inspection, examination and testing personnel.

9a. C.2.a(7)

Requirement

Regulatory Guide 1.58 endorses the guidelines of SNT-TC-1A as an acceptable method of training and certifying personnel conducting leak tests.

Exception/Interpretation

I&M takes the position that the "Level" designation guidelines as recommended in SNT-TC-1A, paragraph 4 do not necessarily assure adequate leak test capability. I&M maintains that departmental supervisors are best able to judge whether engineers and other personnel are qualified to direct and/or perform leak tests. Therefore, I&M does not implement the recommended "Level" designation guidelines.

It is I&M's opinion that the training guidelines of SNT-TC-1A, Table I-G, paragraph 5.2 specifically are oriented towards the basic physics involved in leak testing, and further, towards individuals who are not graduate engineers. I&M maintains that it meets the essence of these training guidelines. The preparation of leak test procedures and the conduct of leak tests at Cook Nuclear Plant is under the direct supervision of Performance Engineers who hold engineering degrees from accredited engineering schools. The basic physics of leak testing have been incorporated into the applicable test procedures. The review and approval of the data obtained from leak tests is performed by department supervisors who are also graduate engineers.

I&M does recognize the need to assure that individuals involved in leak tests are fully cognizant of leak test procedural requirements and thoroughly familiar with the test equipment involved. Plant performance engineers receive routine, informal orientation on testing programs to

ensure that these individuals fully understand the requirements of performing a leak test.

9b. C5, C6, C7, C, C10

Exception/Interpretation

I&M takes the position that the classification of inspection, examination and test personnel (inspection personnel) into "Levels" based on the requirements stated in Section 3.0 of ANSI N45.2.6 does not necessarily assure adequate inspection capability. I&M maintains that departmental and first line supervisors are best able to judge the inspection capability of the personnel under their supervision, and that "Level" classification would require an overly burdensome administrative work load, could inhibit inspection activities, and provides no assurance of inspection capabilities. Therefore, I&M does not implement the "Level" classification concept for inspection, examination and test personnel.

The methodology under which inspections, examinations and tests are conducted at the Cook Nuclear Plant requires the involvement of first line supervisors, engineering personnel, departmental supervisors and plant management. In essence, the last seven (7) project functions shown in Table 1 to ANSI N45.2.6 are assigned to supervisory and engineering personnel, and not to personnel of the inspector category. These management supervisory and engineering personnel, as a minimum, meet the educational and experience requirements of "Level II and Level III" personnel, as required, to meet the criteria of ANSI 1.1 which exceeds those of ANSI N45.2.6. In I&M's opinion, no useful purpose is served by classification of management, supervisory and engineering personnel into "Levels."

Therefore, I&M takes the following positions relative to regulatory positions C5, 6, 7, and 10 of Regulatory Guide 1.58.

C-5 Based on the discussion in 9b, this position is not applicable to the Cook Nuclear Plant.

C-6 Replacement personnel for Cook Nuclear Plant management, supervisory and engineering positions subject to ANSI 18.1 will meet the educational and experience requirements of ANSI 18.1 and therefore, those of ANSI N45.2.6.

Replacement inspection personnel will, as a minimum, meet the educational and experience requirements of ANSI N45.2.6, Section 3.5.1 - "Level I."

C-7 I&M, as a general practice, complies with the training recommendations as set forth in this regulatory position.

C-8 All I&M inspection, examination and test personnel are instructed in the normal course of employee training in radiation protection and the means to minimize radiation dose exposure.

C-10 I&M maintains documentation to show that inspection personnel meet the minimum requirements of "Level I," and that management, supervisory and engineering personnel meet the minimum requirements of ANSI 18.1.

10. N45.2.8,

10a. Sec. 2.9e

Requirement

Section 2.9e of N45.2.8. lists documents relating to the specific stage of installation activity which are to be available at the construction site.

Exception/Interpretation

All of the documents listed are not necessarily required at the construction site for installation and testing. AEPSC and I&M assure that they are available to the site, as necessary.

10b. Sec. 2.9e

Requirement

Evidence that engineering or design changes are documented and approved shall be available at the construction site prior to installation.

Exception/Interpretation

Equipment may be installed before final approval of engineering or design changes. However, the system is not placed into service until such changes are documented and approved.

10c. Sec. 4.5.1

Requirement

"Installed systems and components shall be cleaned, flushed and conditioned according to the requirements of ANSI N45.2.1. Special consideration shall be given to the following requirements: ...." (Requirements are given for chemical conditioning, flushing and process controls.)

Exception/Interpretation

Systems and components are cleaned, flushed and conditioned as determined on a case-by-case basis. Measures are taken to help preclude the need for cleaning, flushing and conditioning through good practices during maintenance or modification activities.

11. N45.2.9

11a. Sec. 5.4, Item 2

Requirement

Records shall not be stored loosely. "They shall be firmly attached in binders or placed in folders or envelopes for storage on shelving in containers." Steel file cabinets are preferred.

Exception/Interpretation

Records are suitably stored in steel file cabinets, or on shelving in containers. Methods other than binders, folders, or envelopes (for example, dividers) may be used to organize the records for storage.

11b. Sec. 6.2

Requirement

"A list shall be maintained designating those personnel who shall have access to the files".

Exception/Interpretation

Rules are established governing access to and control of files as provided for in ANSI N45.2.9, Section 5.3, Item 5. These rules do not always include a requirement for a list of personnel who are authorized access. It should be noted that duplicate files and/or microforms may exist for general use.

11c. Sec. 5.6

Requirement

When a single records storage facility is maintained, at least the following features should be considered in its construction: etc.

Exception/Interpretation

The Cook Nuclear Plant Master File Room and other off-site record storage facilities comply with the requirements of NUREG-000 (7/81), Section 17.1.17.4.

12. Reg. Guide 1.144/ANSI N45.2.12

12a. Sec. C3a(2)

Requirement

Applicable elements of an organization's Quality Assurance Program for "design and construction phase activities should be audited at least annually or at least once within the life of the activity, whichever is shorter."

Exception/Interpretation

Since most modifications are straight forward, they are not audited individually. Instead, selected controls over modifications are audited periodically.

12b. Sec. C3b(1)

Requirement

This section identifies procurement contracts which are exempted from being audited.

Exception/Interpretation

In addition to the exemptions of Reg. Guide 1.144, AEPSC/I&M considers that the National Institute of Standards and Technology; or other State and Federal Agencies which may provide services to AEPSC/I&M, are not required to be audited.

12c. Sec. 4.5.1

Requirement

Responses to adverse audit findings, giving results of the review and investigation, shall clearly state the corrective action taken or planned to prevent recurrence. "In the event that corrective action cannot be completed within thirty days, the audited organization's response shall include a scheduled date for the corrective action."

Exception/Interpretation

AEPSC/I&M take the position that certain circumstances warrant more than thirty (30) days to completely investigate the cause and/or total impact of an adverse finding. For these circumstances, an initial thirty (30) day response will be provided which addresses a schedule for known corrective actions, the reason why additional investigation time is needed, and a schedule for completion of the investigation. These initial responses require the approval of the Director - Quality Assurance.

13. N45.2.13,

13a. Sec. 3.2.2

Requirement

N45.2.13 requires that technical requirements be specified in procurement documents by reference to technical requirement documents. Technical requirement documents are to be prepared, reviewed and released under the requirements established by ANSI N45.2.11.

Exception/Interpretation

For replacement parts and materials, AEPSC/I&M follow ANSI N18.7, Section 5.2.13, Subitem 1, which states: "Where the original item or part is found to be commercially 'off the shelf' or without specifically identified QA requirements, spare and replacement parts may be similarly procured, but care shall be exercised to ensure at least equivalent performance."

13b. Sec. 3.2.3

Requirement

"Procurement documents shall require that the supplier have a documented Quality Assurance Program that implements parts or all of ANSI N45.2 as well as applicable Quality Assurance Program requirements of other nationally recognized codes and standards."

Exception/Interpretation

Refer to Item 2j.

13c. Sec. 3.3(a)

Requirement

Reviews of procurement documents shall be performed prior to release for bid and contract award.

Exception/Interpretation

Documents may be released for bid or contract award before completing the necessary reviews. However, these reviews are completed before the item or service is put into service, or before work has progressed beyond the point where it would be impractical to reverse the action taken.

13d. Sec. 3.3(b)

Requirement

Review of changes to procurement documents shall be performed prior to release for bid and contract award.

Exception/Interpretation

This requirement applies only to quality related changes (i.e., changes to the procurement document provisions identified in ANSI N18.7, Section 5.2.13.1, Subitems 1 through 5). The timing of reviews will be the same as for review of the original procurement documents.

13e. Sec. 10.1

Requirement

"Where required by code, regulation, or contract requirement, documentary evidence that items conform to procurement documents shall be available at the nuclear power plant site prior to installation or use of such items, regardless of acceptance methods."

Exception/Interpretation

Refer to Item 2j.

Requirement

"Post-installation test requirements and acceptance documentation shall be mutually established by the purchaser and supplier."

Exception/Interpretation .

In exercising its ultimate responsibility for its Quality Assurance Program, AEPSC/I&M establishes post-installation test requirements giving due consideration to supplier recommendations.

14. Reg. Guide 1.146/ANSI N45.2.23 and ANSI N45.2.2.12

14a. ANSI N45.2.23, Sec. 1.1

Requirement

This standard provides requirements and guidance for the qualification of audit team leaders, henceforth identified as "lead auditors."

14b. ANSI N45.2.12, Sec. 4.2.2

Requirement

A lead auditor shall be appointed team leader.

Exception/Interpretation

The AEPSC audit program is directed by the AEPSC Director - Quality Assurance and is administered by designated QA Division section managers/supervisor who are certified lead auditors.

Audits are, in most cases, conducted by individual auditors, not by "audit teams." These auditors are certified in accordance with established procedures and are assigned by the responsible QA section manager/supervisor based on their demonstrated audit capability and general knowledge of the audit subject. In certain cases, this results in an individual other than a "lead auditor" conducting the actual audit function.

Established AEPSC audit procedures require that, in all cases, the audit functions of preparation/organization, reporting of audit findings and evaluation of corrective actions be reviewed by QA Division section managers/supervisor, thereby meeting the requirements of ANSI N45.2.23 relative to "lead auditors", and "audit team leaders."

2.2 METEOROLOGY

Due to the extreme importance of site meteorology, particularly with regard to safety considerations, an extensive meteorological study program was initiated at the site during the summer of 1966.

The meteorological features of the plant site were evaluated primarily on the basis of three years data obtained from the 200 foot tower which was installed on the site in 1966. Satellite aerovane stations at inland and on-site locations were used to complement the main tower data. Data from the original meteorological study can be found in the original FSAR.

In most respects, the meteorological patterns are those of a typical open mid-latitude exposure. The wind speeds are strong, variations in direction are frequent and the overall wind rose shows no marked favoritism for any particular direction. The only unusual feature is the low frequency of stable conditions. Both the lapse rate and turbulence class analyses indicate far fewer stable cases than originally anticipated, reaching only 7% over the three year period. Even in the late spring and early summer when the lake is relatively cold, the frequency of stable cases reaches only 20 to 25%. Even more favorable is the very low frequency of the combination of light winds and stable, on-shore flow. Less than 1% of the 200-foot data and only 2.5% of the satellite data are in this category.

The 1992 meteorological data was obtained from the 10 meter level of the main meteorological tower. Supplemental information from the shoreline tower is also considered. Analysis of the 1992 joint frequency tables shows that a small percentage (7%) of the year the stability classification indicated a stable state (Pasquill category G). The combination of stable conditions and on-shore wind occurred only about 1% of the time. Both of these figures correspond very closely with the initial data discussed above.

The only major meteorological hazard expected in the site area is the tornado which has recurrence frequency of over 5000 years at the site itself. Ice storms, which would be expected with greater frequency, are not likely to damage essential facilities, but have been considered in developing certain criteria.

2.2.1 SOURCES OF DATA

Old Site Meteorological Tower

The main source for the inital site meteorological data was a 200-foot meteorological tower which was erected at the site during the summer of 1966 and equipped with meteorological instrumentation (Fig. 2.2-1). This tower remained in continuous operation from October 1966 until 1978. The tower instruments consisted of the following:

200 ft. level - Aerovane and aspirated resistance thermometer.

150 ft. level - Climet Bivane (the extremely strong winds at the site had damaged the Bivane, but some data had been obtained).

50 ft. level - Aerovane and aspirated resistance thermometer.

Ground-level - Resistance thermometer, Dewcell, recording rain gage and recording barometer.

In the Unit 1 Control Room there was instrumentation and a recorder for wind speed, direction and temperature.

2.2.2 GENERAL METEOROLOGY

Southwestern Michigan is typical of the northern lake regions of the United States in most respects. The flat terrain and the frequent passage of well-developed extra-tropical storms create a consistently strong wind flow, as well as rapid changes in both dispersion conditions and wind direction. Some of the meteorological statistics are useful primarily for general planning of the facilities and are therefore reported with a minimum of description. Other data are important in the assessment of safety and these are discussed fully.

<u>High Winds</u>

Strong winds are the most important meteorological hazard to the facilities. The region is frequented by relatively strong, gusty winds, usually accompanying the passage of squall lines or thunderstorms and the maximum wind associated with these phenomena is 90 mph on a 100 year recurrency interval.

The tornado presents a very specialized type of hazard involving both violent winds and extremely large, rapid changes in barometric pressure.

The storms are small, unpredictable in detail and rather infrequent, but they undoubtedly represent one of the few environmental factors that could, if ignored in plant design, inflict direct major damage on the facility. Typically, the tornado is a narrow funnel, often only a few hundred yards wide, in which winds may briefly reach 300 mph. Almost instantaneous changes in barometric pressure occur, reaching 3 psi and causing explosion of vulnerable structures. Because of the severity of the phenomena, very few reliable measurements of tornado intensities exist. It is therefore difficult to dissociate wind and pressure effects, but the estimates given above are considered fairly reliable maximum values. This portion of Michigan has a significant tornado probability, as is apparent in the map shown in Figure 2.2-2. Berrien County has had 25 tornadoes between 1950 and 1989.

Ice Storms

Far less destructive, but far more probable, are the ice storms that frequent the north central states. Michigan lies in the belt where such storms are common and in the years from 1970 to 1989, 6 significant ice storms have been reported in this area.

2.2.3 DISPERSION METEOROLOGY

Worst Case X/Q values

X/Q values for 1992 were calculated from data from the main tower's 10 meter instruments using the MIDAS computer code. The data show a worst case X/Q value as 1.06E-05 sec/m^3 during 1992. The worst case ever computed by the present meteorological system is 1.13E-05 sec/m^3. Both of these values are well below the established worst case X/Q value of 3.15E-04 sec/m^3. Table 2.2-3 shows the X/Q values for all sectors at 10 distances.

Atmospheric Stability

The atmospheric stability for the area is now classified according to the Pasquill categories for use in dispersion calculations. These categories range from A to G, with the G category being the most stable. Joint frequency tables for 1992 have been compiled and are shown in Tables 2.2-4 through 2.2-11. The data show that a large percentage (33%) of the year is devoted to Category D. A rather substantial portion of the year (23%) shows an extremely unstable classification (Category A). There is only a small portion of time (7%) devoted to the extremely stable conditions of Category G.

Wind Speed and Direction

Wind speeds were moderate in 1992. The predominant wind speed range is 4-7 mph category. The wind speed exceeded 14 mph less than 4% of the time. The wind direction at the main tower varied, with the largest

frequencies occuring both from the North and from the South. This can be observed in the wind roses shown in Figures 2.2-3 through 2.2-7. There was a slight tendency for winds from the West (onshore flow) to occur. The second quarter of the year produced winds mostly from the North, while during the fourth quarter they were from the South.

The wind at the shoreline (measured by the shoreline tower) shows a large contrast. The winds are mostly from the South East. This directional preference can be seen in all four quarters of the year, as shown on the wind roses in Figures 2.2-8 through 2.2-12.

REFERENCES, SECTION 2.2

1. Fawbush, Miller and Starrett: An Empirical Method of Forecasting Tornado Development, Bulletin, AMS, 32, 1951.

2. Spohn et. al.: Tornado Climatology, Monthly Weather Review, Wash., D.C., 1962.

3. Thom: Tornado Probabilities, Monthly Weather Review, Wash., D.C., 1963.

4. Thom: Distributions of Extreme Winds in the United States, Journal, Struct. Div. ASCE, April, 1960.

5. Singer and Smith: Relation of Gustiness to Other Meteorological Variables, Journal of Met., 1953.

6. Michigan Emergency Management Division, Michigan Hazard Analysis, September 1992.

2.7 ENVIRONMENTAL RADIATION MONITORING

The radiological environmental radiation monitoring program, described herein, was designed to evaluate the effects that routine and inadvertent radioactive releases from the plant have on the environment. Provisions are made to monitor liquid and gaseous wastes before and/or during their release, and further to monitor the atmosphere, lake water, well water, aquatic organisms, milk and food materials when necessary. Liquid and gaseous wastes are released as continuous releases as well as batch releases from time to time, after appropriate decay, processing and analysis. No foreseeable environmental conditions will restrict the release of wastes. However, if extreme conditions should indicate the desirability, wastes can be retained until dispersion conditions improve.

2.7.1 DETERMINATION OF MAXIMUM ALLOWABLE RELEASE RATES TO AIR AND WATER

The first step in the program is to determine the maximum rate at which radioactive material may be continuously discharged without exceeding the limits set by 10 CFR 20 at the site boundary. The initial estimate is based on the plant design, the anticipated composition of the radioactive material to be released, and the dilution and dispersion characteristics of the air and water into which these materials are discharged.

The unit vent, through which radioactive gaseous waste is routinely released, runs up the outside wall of the containment building. The vent opening is at the top of the containment building, about 160 feet above grade.

Other release pathways for discharge of radioactive gaseous waste include the steam jet air ejectors, gland seal leak-offs, steam generator blowdown, and the auxiliary boiler stack.

Table 2.2-11 shows the predominant wind to be coming from the South. The values calculated for X/Q for all of the wind directions are given in Table 2.2-3. Using the table, the X/Q for winds blowing toward the North (from the South) is calculated to be 9.54E-06 seconds/m^3. For the waste gas mixture in Table 11.1-6, the weighted average maximum permissable concentration is 1.2E-07 μCi/cc. The maximum rate at which this mixture can be discharged continuously is Q = .013 Ci/sec, without exceeding the 10 CFR 20 limit at the site boundary.

If suitable for discharge, liquid radioactive wastes are released to the condenser circulating water system. The maximum discharge concentration for liquids is defined in the Plant Technical Specifications. This defined concentration ensures that the limits established in 10 CFR 20 are not violated.

2.7.2 SAMPLING STATIONS

The stations for sampling airborne particulates, volatiles, and external radiation are placed in two rings about the plant. The inner, or indicator ring, stations are placed where it is estimated that maximum ground concentrations of material released from the plant will occur. Figure 2.7-1 indicates the locations selected for the six indicator stations (shown as A1 through A6).

Figure 2.7-3 shows the locations which have been selected for the four background air stations in the outer ring as identified as A. These locations are all about 20 miles from the plant and thus the ground-level concentrations of radioactive material originating from the plant will be less than 1 percent of the concentrations at the indicator stations.

Locations of TLD stations are shown in Figures 2.7-1, 2.7-3 and 2.7-4. Twelve on-site indicator TLD stations (shown as A1 through A12 on Figure 2.7-1) are located on an approximate 2000 foot radius and eleven off-site monitoring TLD stations are within a 2 to 5 mile radius from the plant (shown as T1 through T11 on Figure 2.7-4). Four background TLD stations located about 20 miles from the plant are identified as A on Figure 2.7-3.

Sampling Lake Water

The locations of the sampling stations for lake water are described in Table 2.7-1. Indicator lake samples are taken along the lake front from the condenser cooling water intake and at an approximate distance of 0.1 and 0.2 miles north and 0.1 and 0.3 miles south of the plant centerline.

The sampling of aquatic organisms presents a number of difficulties. Out to a depth of 20 feet or more, the lake bottom is scoured sand and is almost sterile. Attempts to find suitable organisms in sufficient quantities for routine sampling have been unsuccessful.

Benthonic organisms occur only at depths greater than twenty feet; such depths occur at 1,000 feet or more from shore. Routine sampling under such circumstances is impractical for extended periods of unfavorable weather.

Fish are collected and analyzed in the program, but fish are a poor sampling medium because they range so widely that it is never certain that they represent the area where they happen to have been caught.

Sampling of Well Water

Well water is the only material in the environmental sampling program that is not likely to be affected by fallout of radioactivity. With well water, and only with well water, is the before and after principle sound. There are 17 wells (13 REMP wells and 4 non technical specification steam generator storage facility groundwater monitoring wells) within the owner controlled area as shown in Tables 2.7-2 and 2.7-3. The orientation of these wells with respect to the plant was chosen as a result of groundwater movement, which was found to be east to west.

Sampling of Milk

The selection of milk sampling locations are, of course, limited to pastures where milk cows graze. The locations shown in Figure 2.7-3 are subject to change as the location of milk cows change.

Sampling of Food

It is now evident that milk alone provides sufficient control of terrestrial pathways. Additional human food materials are not needed in the program unless radioactive materials other than noble gases, tritium and iodine are detected in the plant discharges to the atmosphere. Because radioactive particulates have been noted in gaseous discharges additional human food crops will be sampled annually.

The noble gases do not enter directly into the food chains. Tritium enters freely into all food chains; however, since almost all tritium occurs as tritiated water, it does not concentrate in food pathways as do other elements. Iodine does concentrate along food pathways and it has been shown that the air-pasture-milk animal-milk pathway is critical and that milk is the best monitoring medium. Radioactive particulates which settle out on the surfaces of crops are adequately monitored by the sampling and analysis of broadleaf vegetation and grapes. There is, consequently, neither need nor justification for monitoring human foods other than milk and selected vegetation in the terrestial environment, and fish in the aquatic environment.

All sampling points have been selected because they are representative of the area and accessible for sampling. Table 2.7-5 describes the current Radiological Environmental Monitoring Program, as defined in the plant Technical Specifications.

2.7.3 STABLE ELEMENT STUDIES

The pre-operational phase of the environmental program includes a study of stable element concentrations in the lake water and in selected aquatic organisms. The purposes of these measurements are (1) to put an upper limit on the degree to which radioactive material discharged from the plant into the lake could be concentrated in human food taken from the lake, (2) to find critical pathways and the means for estimating population exposure by these pathways, and (3) to determine

the relationship between the concentration factors in fish (and any other human foods taken from the lake) to those in aquatic organisms selected to monitor the water environment.

The principle involved in these stable element studies is that the radioactive isotopes of an element cannot be concentrated more highly than the corresponding stable isotopes of that element by biological, chemical or physical processes in the environment. The general form of these studies is described in the next paragraph.

The radioactive isotopes anticipated in the liquid waste (Table 11.1-5) are examined, as are the data on similar operating reactors. From these one obtains a list of the elements which correspond to all the radioactive isotopes which may contribute to radioactivity in food chains. Samples of lake water, edible portions of fish, and other possible monitoring organisms, if available, are collected and analyzed for each of the elements in the list. The data so obtained give concentration factors from water to fish, and from water to monitor organisms for the stable elements. Radioactive isotopes of these elements cannot be concentrated to factors greater than those for the corresponding stable elements.

2.7.4 MEASUREMENT OF RADIOACTIVITY

The pre-operational phase of the environmental program included the collection and analysis of samples for radioactivity; the intensity of the post-operational phase is concerned exclusively with radioactivity released from the plant. This section describes the equipment and techniques that are used to collect and analyze environmental samples for radioactivity.

Direct radiation doses primarily due to radioactive noble gases in the environment is measured with thermoluminescent dosimeters. The detection limit of thermoluminescent dosimeters is 1 to 2 mR per month. This

sensitivity corresponds to 2 to 4 percent of the maximum permissible dose to the public from radioactive noble gases.

The air sampling units draw about 6×10^{8} cc of air per week through a filter. The detection limit of a lithium drifted germanium gamma spectrometer is on the order of 10^{-9} Ci/cc, far below the maximum permissible public concentration of any radioactive material the plant could discharge to the atmosphere.

The environmental air sampling units are fitted with charcoal cartridges to collect iodine. If the air passing through this cartridge were at the public maximum concentration of iodine 131 for the entire week, the cartridge would collect 0.06 μCi.

Tritium is measured in a liquid scintillation counter with a nominal sensitivity limit of 10^{-4} μCi/cc, which is 0.03 of the permissible drinking water concentration. Analyses will be made by contract with an outside laboratory.

2.7.5 OPERATION OF THE PROGRAM

The environmental radiation monitoring program was started some 12 to 18 months before fuel was loaded in Unit 1. During this period, equipment was tested, the suitability of the selected sampling media and sample points were determined, analytical procedures were tested, and some data was accumulated and examined for statistical variability. Modifications that were necessary to attain reliable and coherent data were made during this period.

Prior to any liquid or gaseous release, the concentration of radioactive material that is to be released to the environment is determined. Based on the total concentration of radioactive material present, the release flow

rate is adjusted to ensure that the Technical Specification release limits are not exceeded. In addition the actual dose to a member of the public resulting from liquid and gaseous releases from the Cook Nuclear Plant is determined to ensure that Technical Specification requirements are not being exceeded.

2.7.6 SUMMARY OF PREOPERATIONAL ENVIRONMENTAL MONITORING PROGRAM

The average monthly LiF Dosimeter Loadings for the quarter of August, 1971 through December, 1971, on site, vary from 3.9 ± 1.3 to 11.7 ± 0.8 mrem and offsite 3.9 ± 1.2 to 13.3 ± 1.1 mrem.

Initial water samples taken in the Lake (similar to that shown in Figure 2.7-1) and at Bridgman, St. Joseph, Benton Harbor and New Buffalo show a tritium concentration of from 0.562 ± 0.036 to 0.583 ± .036 picocuries/milliliter. Gross beta at the above sampling points showed 0.0 ± 2.0 picocuries/liter to 6.8 ± 1.0 picocuries/liter.

The determination of gross beta in the air particulates on site is 0.01 ± 0.01 to 0.30 ± 0.01 picocuries/cubic meter. The same values for offsite stations are 0.01 ± 0.01 to 0.24 ± 0.1 picocuries/cubic meter.

TABLE 2.7-1

LOCATIONS OF THE WATERBORNE SURFACE SAMPLING STATIONS

Indicator Stations

Condenser cooling water intake (L1).

0.3 miles southwest from plant centerline along the lake shore (L2).

0.2 miles northeast from plant centerline along the lake shore (L3).

0.1 miles southwest from plant centerline along the lake shore (L4).

0.1 miles northeast from plant centerline along the lake shore (L5).

(See Figure 2.7-1)

Background Stations (and drinking water sample stations)

Lake Township water intake, 0.4 miles south from the plant (D_A).*

St. Joseph municipal water intake, 9 miles northeast from the plant (D_B).*

(See Figure 2.7-3)

*D_A and D_B refer to analysis performed as indicated in Table 2.7-4.

TABLE 2.7-2

WELLS AVAILABLE FROM MONITORING PROGRAM
(Refer to Figure 2.7-1 for a map indicating
the location of these sample points)

Well No.	Approximate Distance from Plant in Feet	Direction from North
W1	1969	11°
W2	2292	63°
W3	3279	107°
W4	418	301°
W5	404	290°
W6	424	273°
W7	1895	189°
W8	1279	53°
W9	1447	22°
W10	4216	129°
W11	3206	153°
W12	2631	162°
W13	2152	182°

TABLE 2.7-3

NON TECHNICAL SPECIFICATION GROUNDWATER WELLS

STEAM GENERATOR STORAGE FACILITY

(Refer to Figure 2.7-2 for a map indicating the location of these sample points)

Well No.	Approximate Distance from Plant in Feet	Direction from North
SGR-1	4037	95°
SGR-2*	3879	92°
SGR-4	3699	93°
SGR-5	3649	92°

These wells are sampled and analyzed quarterly for:

- Gross alpha Activity
- Gross Beta Activity
- Gamma Isotopic Activity

*No SGR-3 well defined for this program.

Table 2.7-4 intentionally deleted.

TABLE 2.7-5

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
1. Airborne a. Radioiodine and Particulates	A1-A6 (Site) New Buffalo, South Bend, Dowagiac, and Coloma are Background	Continuous operation of sampler with Sample Collection as required by Dust Loading But at Least Once Per 7 Days	Radioiodine canister Analyze: Weekly for I-131 Particulate sample Gross Beta Radio-activity following Filter Change[a], composite (by location) for gamma isotopic quarterly.
2. Direct Radiation	a) A1-A12 (On-Site) b) New Buffalo, South Bend, Dowagiac, Coloma c) 11 Off-Site TLD Monitor Locations	At least once per 92 Days (Quarterly)	Gamma Dose. At Least Once Per 92 Days.
3. Waterborne a. Surface	L1, L2, L3, L4, L5	Composite* Sample Over One-Month Period	Gamma Isotopic Analysis monthly. Composite for tritium analysis-quarterly.

*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 24 hours.

[a]Particulate sample filters should be analyzed for gross beta 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air or water is greater than 10 times the yearly mean of control samples for any medium, gamma isotopic analysis should be performed on the individual samples.

TABLE 2.7-5 (Cont'd)

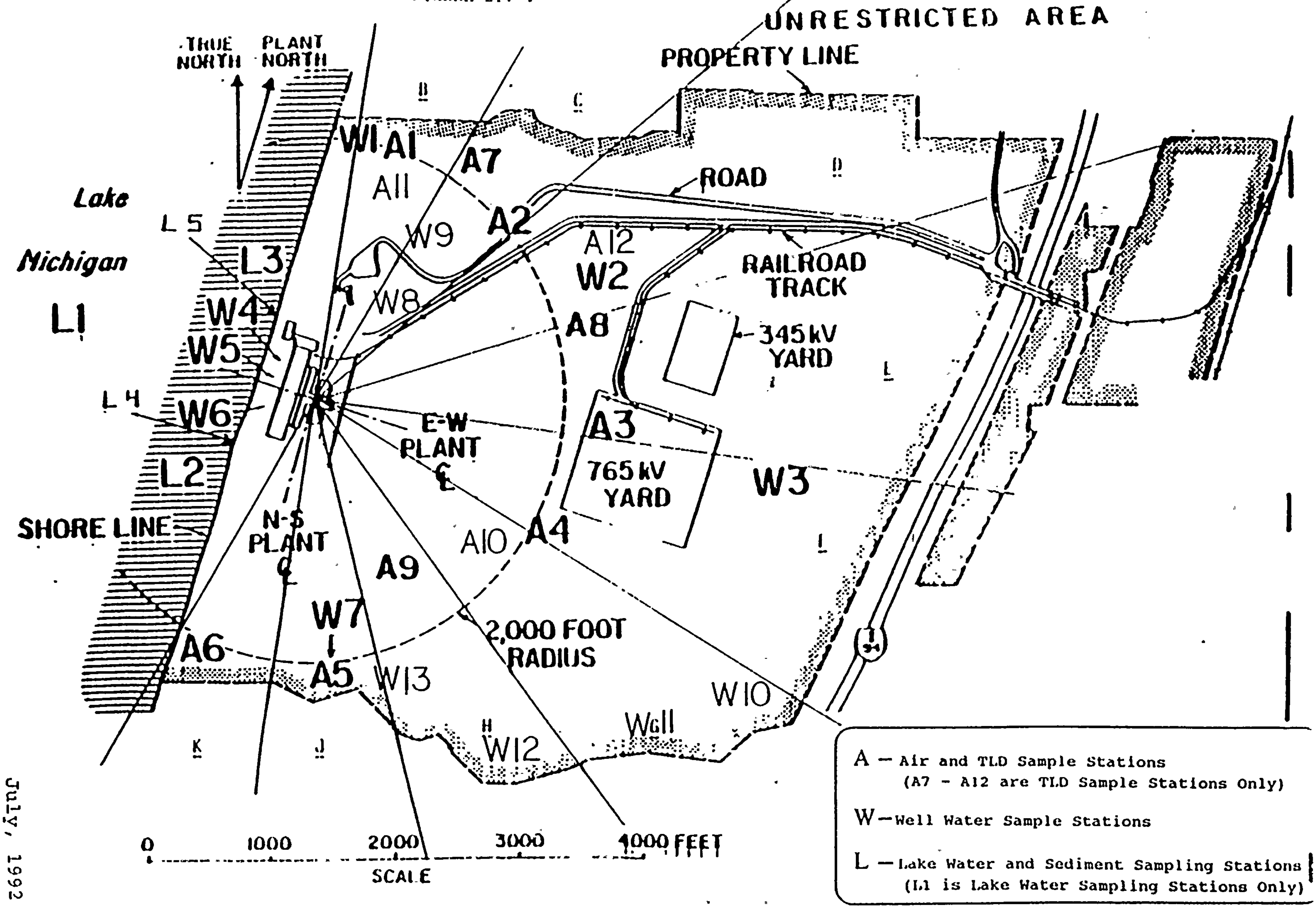
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
b. Ground	W1-W13	Quarterly	Gamma Isotopic and Tritium analysis quarterly.
c. Drinking	St. Joseph Lake Township	Composite* Sample Collected over a Period of less than or equal to 31 days Composite* Sample Over a 2-week Period if I-131 Analysis is Performed.	Gross Beta and Gamma Isotopic Analysis of each composite sample. Tritium Analysis of composite Quarterly. I-131 analysis on each composite when the dose calculated for the consumption of the water is greater than 1 mrem per year.
d. Sediment from Shoreline	L2, L3, L4, L5	2/year	Gamma Isotopic Analysis Semi-Annually.
4. Ingestion a. Milk	Indicator Farms** Background Farms	At Least Once Per 15 Days When Animals are on Pasture. At Least Once Per 31 Days at Other Times.	Gamma Isotopic and I-131 Analysis of Each Sample.

*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 24 hours.

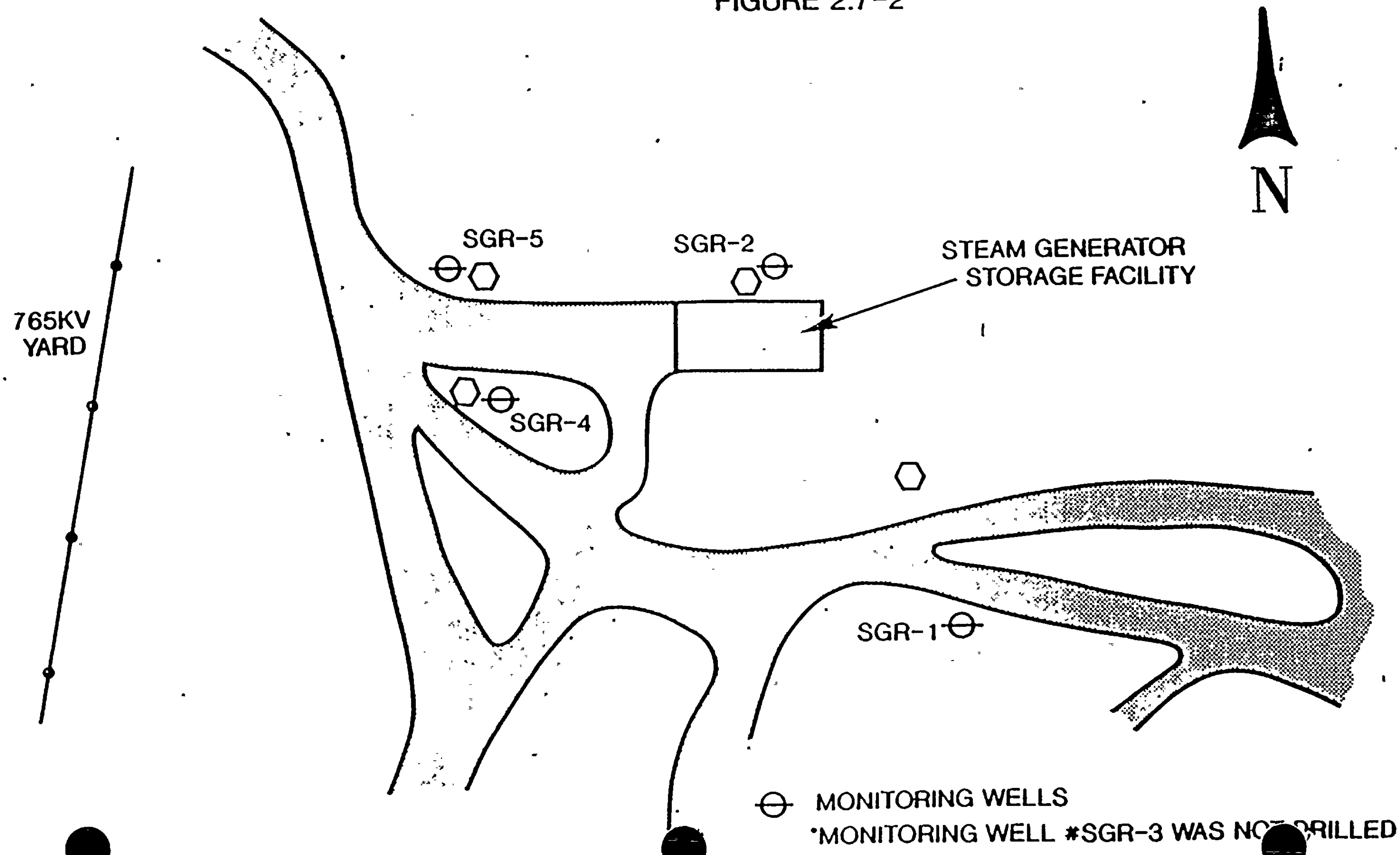
**An indicator farm is defined as the nearest milk producer in each of the land sectors within 8 miles of the plant site who is willing to participate in the radiological environmental monitoring program. A background farm is defined as a milk producer in one of the less prevalent wind directions at a distance greater than 15 miles but less than 25 miles who is willing to participate in the radiological environmental monitoring program. If at least three indicator milk samples and one background milk sample cannot be obtained, vegetation sampling will be performed as a replacement for the milk sampling and no milk samples will be required.

FIGURE 2.7-1


UNRESTRICTED AREA
PROPERTY LINE
TRUE NORTH
PLANT NORTH
Lake Michigan
L1
L2
L3
L4
L5
W1
W2
W3
W4
W5
W6
W7
W8
W9
W10
W11
W12
W13
A1
A2
A3
A4
A5
A6
A7
A8
A9
A10
A11
A12
ROAD
RAILROAD TRACK
345 kV YARD
765 kV YARD
E-W PLANT ℄
N-S PLANT ℄
SHORE LINE
2,000 FOOT RADIUS
B
C
D
L
H
J
K
0
1000
2000
3000
4000 FEET
SCALE
A — Air and TLD Sample Stations (A7 - A12 are TLD Sample Stations Only)
W — Well Water Sample Stations
L — Lake Water and Sediment Sampling Stations (L1 is Lake Water Sampling Stations Only)

STEAM GENERATOR STORAGE FACILITY
NON-TECHNICAL SPECIFICATION
GROUNDWATER MONITORING WELLS

FIGURE 2.7-2



July 1989

In the dynamic piping analyses, vertical seismic spectra equal to 2/3 of the pertinent building base horizontal spectra was computer input with the appropriate building floor horizontal seismic spectra. The effects of each seismic spectra input were computed independently and the various modal results were computer combined by the square root of the sum of the squares (SRSS) method. The effects of the vertical and a horizontal seismic run were then computer combined by the SRSS method. The larger resultant value of the vertical and horizontal seismic run [(Y + X), or (Y + Z)] at each node was considered to be the critical load and/or stress.

Class I piping smaller than 2½ inch nominal diameter with operating temperatures less than 250°F, may be qualified by using either a simplified analysis (Alternate Analysis) method or a computer dynamic analysis. The Alternate Analysis method developed for the Cook Nuclear Plant considered gravity loads, seismic loads (based on floor acceleration response spectra) and internal pressure loads. The acceptance criteria were based on pipe stress and pipe displacement. A set of instructions, guidelines, tables and graphs reflecting the above, were issued to establish acceptable spacing of supports.

Class II piping with operating temperatures less than 250°F may be qualified by using this Alternate Analysis method. Class II piping with operating temperatures greater than 250°F are qualified using the computer dynamic analysis method. The seismic inputs are taken from the appropriate OBE spectra.

Where a piping system consists of a combination of Class I and/or Class II, and/or Class III piping, the method of analysis is for the higher class piping. The piping model maybe structurally decoupled, to suit the higher class piping, at an anchor or at a point (or points) encompassing restraints in the 3 orthogonal directions.

2.9.4 SEISMIC DESIGN CRITERIA FOR CLASS I, CLASS II AND CLASS III STRUCTURES

Class I

A dynamic analysis was performed using Response Spectrum and Modal Analysis Procedure, as discussed in Appendix F of the original FSAR, "Dynamic Analysis of the Containment Structure for Seismic Loading." Response spectra were generated from information obtained by a full seismological study of the site. Stress criteria are those of ACI 318-63 Ultimate Strength Design.

Class II

An analysis using the procedures of the Uniform Building Code (International Conference of Building Officials) was made. Standard working stresses are used.

Values of maximum ground acceleration are those used for Class I criteria. The factor applied to the seismic forces from which the values of shear, bending moments, etc. are computed, is taken as that for Zone 3 of the Uniform Building Code multiplied by the ratio of the maximum ground acceleration to a value of 0.30g. The minimum ratio used is one-fourth.

Class III

An analysis using the procedures of the Uniform Building Code (International Conference of Buiilding Officials) was made. Standard working stresses increased by 33 percent are used. Zonal factors of the Uniform Building Code are used.

For All Structure Seismic Classifications

A vertical component of earthquake acceleration of two-thirds the value of the horizontal component of earthquake is assumed to be acting simultaneously with the horizontal component.

Seismic design criteria for combined structures (i.e., structures having Class I and Class II elements, Class I and Class III elements or Class II and Class III elements) are as follows:

1. Equipment is supported by structural elements equal to or higher than the classification of the equipment.

2. Equipment is surrounded by structural elements equal to or higher than the classification of the equipment.

3. Structural elements are supported by, or framed to, elements equal to or higher than its own classification.

The following example illustrates the design criteria stated above.

The auxiliary feed pumps are Class I equipment but are housed in the turbine building which is essentially a Class III structure. In this case, the Class I equipment is anchored directly to the foundation slab which is designed to Class I criteria. The pumps are surrounded by local structural elements designed to Class I criteria which have been designed to withstand potentially adverse effects of lower class structures in the area.

The superstructure for the turbine room, heater bay and main steam pipe enclosure beyond the steam generator stop valve are Class III structures, which are designed for seismic loading in accordance with the seismic criteria of the Uniform Building Code. The maximum deflection for all conditions of loading were computed for these structures.

These deflections plus an allowance for erection and fabrication tolerances and an additional amount for clearance were designed into these structures to prevent rattling (hammering) effect.

The primary water and condensate tanks are functionally Seismic Class II structures located near Seismic Class I structures, namely, the refueling water storage tank and the containment. The condensate and refueling water storage tanks have been seismically analyzed to insure their structural integrity during a seismic event. The primary water tank was analyzed seismically for the OBE. All three tanks are located in excess of 20 feet from the containment wall. The primary water storage tank is approximately 55 feet from the refueling water storage tank. Analysis indicates that the primary water storage tank will not cause structural damage to the refueling water or condensate storage tanks in the unlikely event that it fails. The condensate storage tank, although not required to be a Seismic Class I structure, was designed as such to insure the structural integrity of the refueling water tank.

2.9.5 GENERAL DESIGN CONSIDERATIONS FOR BUILDING STRUCTURES

Those structures considered are the auxiliary, containment, circulating water pump screen house and turbine buildings, and the steam generator stop valves and pipe enclosures outside the containment building.

Building structures were designed to withstand wind forces.

Class I building structures were evaluated with reference to tornado conditions to assure that there would be no loss of function.

The wind velocities and tornado model are discussed in Chapter 5 and Subchapters 1.4 and 2.8.

Tornado loading was not considered coincident with earthquake loading. However, a 3 psi ambient pressure drop was considered coincident with tornado velocity pressures.

Pressure and suction forces together with internal pressure or suction was considered in accordance with the procedure in ASCE Paper No. 3269 "Wind Forces on Structures."

Torsional effects due to tornado loading were considered in evaluating Class I structures.

Maximum torsional loading was determined by using varying diameter tornado "funnels."

Reinforcing was placed so that minimum reinforcing cover provisions are as recommended by the Uniform Building Code and ACI Building Code.

1. 3 in. cover where concrete was deposited against the ground (bottom of slab).

2. 2 in. cover at all formed surfaces exposed to the ground or weather (all exterior surfaces of the structure).

3. 1½ in. cover for beams and girders not exposed to the ground or the weather.

4. 1 in. cover for slabs and walls not exposed to the ground or weather.

5. Concrete protection for reinforcement is in all cases at least equal to the diameter of the bars.

Building structures were designed in accordance with the seismic design criteria as stated in Section 2.9.4.

The effects of differential motion between the various buildings were considered. This was necessary both to provide adequate separation between the structures to prevent "banging together" of the structures during a seismic occurrence and to provide for this condition on interconnecting elements.

Both the horizontal and rotational motions of the containment structure due to earthquake were analyzed. A plot of displacement vs height was made.

The magnitude of maximum vertical motion due to the DBE was determined for the structures, considering each structure as a rigid body.

The maximum magnitude of differential motion was considered to be the absolute value of the peaks of motion between the independent structures, considering each motion to occur simultaneously with the others.

The effect of static differential settlement was considered additive to the dynamic effects where this resulted in a more severe condition.

A discussion of the design for the auxiliary and turbine building follows. The design of the containment building is discussed in Chapter 5.

Auxiliary Building

The Auxiliary Building encloses the fuel storage areas (both new and used fuel), the fuel transfer canal, the containment equipment hatches access areas, control facilities and other equipment.

Seismic considerations for the Auxiliary building were based on the 10% and 20% Ground Response Curves as indicated in Figures 2.5-2 and 2.5-3. A dynamic analysis of the building was performed to determine the seismic stresses in Class I portions of the structure. Using a slab-spring model

subjected to independent translational excitation in two perpendicular directions, the modal periods, the forces acting on the slabs, the slab displacement and the loads on major lateral load resisting elements were computed. Consideration was also given to the action of water in the spent fuel pool during a seismic occurrence.

The superstructure is a Class I structure consisting of a structural steel skelton with exterior walls and roof of reinforced concrete.

The structural steel was designed in accordance with the "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," adopted April 17, 1963, by the American Institute of Steel Construction.

The roof of the structure is constructed of steel beams and girders supporting a poured concrete roof on steel ribbed decking. The roof varies in thickness, stepped from two feet to seven inches. The thickest portion of the roof is directly over the spent fuel pool area.

The walls are of poured concrete supported, for their vertical load, on the concrete substructure and for their lateral forces by the structural steel columns and struts. The walls vary in thickness from two feet to six inches. The thickest area is the west wall adjacent to the fuel pool and the thinnest portion is at the east end of the structure.

The concrete walls and roof of the auxiliary building were designed to provide protection against potential missiles. The whole structure was designed to withstand the design basis tornado missiles and was also designed to protect the control room and fuel pool against a turbine missile. See Sub-Chapter 1.4 for a discussion on missile protection.

The tornado forces applied to the structure are as outlined in Chapter 5 with the exception that the diameter of the tornado was assumed to vary in the following manner:

a. The diameter is equal to the width of the structure.

b. The diameter is equal to the length of the structure.

c. The diameter is infinite in extent.

In the event of a tornado, the pressure within the structure will not differ from the outside by more than 1/2 psi in three seconds. This low differential is achieved by the installation of vents in the periphery of the roof which will allow release of internal pressure. However, as an added conservatism, the building roof and walls have been designed to withstand 3/4 psi coincident with tornado wind forces. For forces resulting from tornado winds of 250 mph tangentially and a progression velocity of 50 mph, the auxiliary building steel will not experience stresses in excess of allowable as outlined in the 1963 American Institute of Steel Construction specifications. For tornado winds of 300 mph tangentially with a progression of 60 mph, coincident with internal pressures of 3/4 psi, steel will remain within yield and no permanent deformation will result.

The auxiliary building, as a Class I structure, has been designed for seismic forces as described in this chapter. A dynamic analysis was made for the OBE and DBE. For the OBE, all stresses in the steel superstructure are within allowables as specified by the 1963 code of the "American Institute of Steel Construction for Buildings." For the DBE, the superstructure steel stresses do not exceed yield and no permanent deformations will result.

For the 1988 Steam Generator Replacement Project, the following changes were made to the auxiliary building. An additional 150 ton single failure proof crane was installed and the existing 150 ton crane was upgraded to a single failure proof design. The building was reanalyzed for the following conditions:

a. The two cranes acting in tandem to move steam generator components during the replacement project.

b. The seismic forces, as described in this chapter, resulting from a single crane with a 60 ton live load acting anywhere in the building.

For the OBE, all stresses in the steel superstructure are within allowable as specified by the code of the "American Institute of Steel Construction for Buildings." Adopted November 1, 1978. For the DBE, the superstructure steel stresses do not exceed yield and no permanent deformation will result.

Turbine Building

A structure such as the turbine building, which consists of a Class I foundation and a superstructure which is Class I in some areas and Class III in other areas, was designed as follows. The superstructure was designed in accordance with the criteria discussed in Section 2.9.4. The reactions at the base of the superstructure were used as input for the foundation design. The foundation was analyzed for lateral earthquake and a simultaneously acting vertical component, considering the effects on the foundations of the superstructure and any equipment supported directly on the foundation.

For seismic or tornado conditions, the mat was designed in accordance with the stress criteria of ACI Code 318-63 "Ultimate Strength Design". The load equations used were those of Subsection 5.2.2.2, with the elimination of the pressure and temperature items.

For normal load conditions, the mat was designed using Working Stress Design. Stresses and strains for normal loading were held to the limits of ACI Code 318-63 "Working Stress Design."

In the design of Class I structures by ACI Code 318-63 "Ultimate Strength Design" procedure, load reduction factors (Ø) used for the containment are discussed in Subsection 5.2.2.2. However for structures other than the

containment structure and when considering seismic conditions, the load reduction factor for diagonal tension, bond and anchorage in concrete was reduced to 0.75.

2.9.6 SEISMIC DESIGN CRITERIA FOR EQUIPMENT

Seismic Class I equipment design generally requires that normal plus DBE stresses do not exceed yield, and rotating or sliding equipment functions do not bind. The combination of earthquake plus normal stresses for the OBE condition shall not exceed normal allowable, as defined by applicable code. Refer to Table 2.9-1, and Notes thereto, for the definition of loading conditions.

Restraints for both Class I mechanical and electrical equipment were generally designed to accept combined normal plus DBE loading without exceeding 0.9 of the yield stresses.

Class I equipment was designed for earthquake loads represented by the combination of appropriate horizontal and vertical floor responses simultaneously applied. The vertical response was equal to 2/3 of the horizontal response.

Depending on the relative structural complexity and relative rigidity of the equipment to be evaluated, one of the following methods of seismic qualification was performed:

1. For structurally complex equipment, a dynamic multi-degree-of-freedom modal analysis which considered frequency, mode shape and modal participation factors in determining seismic response.

2. For structurally simple equipment, a dynamic single degree-of-freedom analysis which considered fundamental frequency response of the equipment as determined from the floor response spectrum.

3. A simplified dynamic analysis which utilized the peak of the floor response spectrum to determine seismic loading.

4. Testing of identical or similar components using approved procedures to simulate appropriate seismic loads.

TABLE 2.9-1

LOADING CONDITIONS

CONDITIONS*	EFFECTS CONSIDERED
1. NORMAL	Deadweight, Thermal, Pressure (Pressure is considered for vessel and pipe stress only)
2. UPSET	Same as 1, OBE
3. EMERGENCY	Same as 1, DBE (Thermal is considered for supports loads only)
4. FAULTED	Same as 1, Postulated Pipe Rupture (Thermal is considered for supports loads only)
5. FAULTED (Including DBE)	Same as 3, Postulated Pipe Rupture

* See Note 1.

TABLE 2.9-1

NOTES

NOTE 1: Definition of Terms based on the Summer 1968 Addenda to the ASME Boiler and Pressure Vessel Code, Section III.

The Operating Load Combination categories are defined as follows:

(1) Normal Condition - Any condition in the course of system startup, operation in the design power range and system shutdown, in the absence of Upset, Emergency or Faulted Conditions.

(2) Upset Condition - Any deviations from Normal Conditions anticipated to occur often enough that design should include a capability to withstand the conditions without operational impairment. The Upset Condition includes those transients caused by a fault in a system component requiring its isolation from the system, transients due to a loss of load or power and any system upset not resulting in a forced outage. The Upset Conditions include the effect of the specified earthquake for which the system must remain operational or must regain its operational status.

(3) Emergency Condition - Any deviations from normal conditions which require shutdown for correction of the conditions or repair of damage in the system. The conditions have a low probability of occurrence but are included to provide assurance that no gross loss or structural integrity will result as a concomitant effect of any damage developed in the system. The total number of postulated occurrences for such events shall not exceed twenty-five (25). Among the Emergency Conditions may be a specified earthquake for which safe shutdown is required.

TABLE 2.9-1

(4) Faulted Condition - Those combinations of conditions associated with extremely low probability postulated events whose consequences are such that the integrity and operability of the nuclear energy system may be impaired to the extent where considerations of public health and safety are involved. Such considerations require compliance with safety criteria as may be specified by jurisdictional authorities.

TABLE 2.9-2

(Part A)

LOADING CONDITIONS AND STRESS LIMITS: PRESSURE VESSELS

LOADING CONDITIONS	STRESS INTENSITY LIMITS	NOTE
1. Normal Conditions	(a) $P_m \leq S_m$	
	(b) P_m (or P_L) + $P_B \leq 1.5S_m$	1
	(c) P_m (or P_L) + P_B + $Q \leq 3.0S_m$	2
2. Upset Condition	(a) $P_m \leq S_m$	
	(b) P_m (or P_L) + $P_B \leq 1.5S_m$	1
	(c) P_m (or P_L) + P_B + $Q \leq 3.0S_m$	2
3. Emergency Condition	(a) P_m $1.2S_m$ or S_y whichever is larger	
	(b) P_m (or P_L) + $P_B \leq 1.5(1.2S_m)$ or $1.5S_y$ whichever is larger	3
4. Faulted Condition	See Note 4	

P_m - primary general membrane stress intensity

P_L - primary local membrane stress intensity

P_B - primary bending stress intensity

Q - secondary stress intensity

S_m - stress intensity value from ASME B&PV Code, Section III, Nuclear Vessels - 1968 Edition, Table N-421

S_y - minimum specified material yield (ASME B&PV Code, Section III, Nuclear Vessels - 1968 Edition, Table N-424

TABLE 2.9-2

(Part B)

LOADING CONDITIONS AND STRESS LIMITS: PRESSURE PIPING

LOADING CONDITIONS	STRESS LIMITS
1. Normal Conditions	(a) $P_m \leq S_h$
	(b) $P_L + P_B \leq S_h$
	(c) $S_A \leq (1.25S_C + 0.25S_h)\ f$
2. Upset Conditions	(a) $P_m \leq 1.2S_h$
	(b) $P_L + P_B \leq 1.2S_h$
	(c) $S_A \leq (1.25S_C + 0.25S_h)f$
3. Emergency Conditions	(a) $P_m \leq 1.2S_h$
	(b) $P_L + P_B \leq 1.5(1.2S_h)$
4. Faulted Conditions	See Note 4.

where:

P_m - primary hoop membrane stress (pressure)

P_L - primary longitudinal membrane stress (pressure)

P_B - primary longitudinal bending stress (deadweight, seismic)

S_h - allowable stress at temperature from USAS B31.1 Code for Pressure Piping, 1967 Edition

S_C - allowable stress at 70°F from USAS B31.1 Code for Pressure Piping, 1967 Edition

S_A - allowable stress range for expansion stresses (fatigue criteria)

f - stress range reduction factor for cycling per USAS B 31.1 Code for pressure piping, 1967 Edition

TABLE 2.9-2

(Part C)

LOADING CONDITIONS AND STRESS LIMITS: EQUIPMENT SUPPORTS

LOADING CONDITIONS	STRESS INTENSITY LIMITS
1. Normal Condition	Working Stresses or Applicable Factored Load Design Values
2. Upset Condition	Working Stresses or Applicable Factored Load Design Values
3. Emergency Condition	Within yield after load redistribution
4. Faulted Condition	Permanent Deflection of Supports Limited to Maintain Supported Equipment Within Design Limits. See Note 4

Support loads are combined by algebraic summation, in plus and minus directions of the three orthogonal planes, so as to obtain the maximum positive or maximum negative value of design load.

The thermal load component is not considered when algebraic summation with this load would lessen the support design load.

The seismic load component is considered to have both a positive and a negative sign. The sign of the seismic component is chosen so as to maximize the absolute value of the support design load.

TABLE 2.9-2

NOTES

Note 1: The limits on local membrane stress intensity ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) + $P_B \leq 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings do not exceed 2/3 or the lower bound collapse load as per paragraph N417.6(b) of the ASME B&PV Code, Section III, Nuclear Vessels - 1968 Edition.

Note 2: In lieu of satisfying the specific requirements for the local membrane ($P_L \leq 1.5S_m$) or the primary plus secondary stress intensity ($P_L + P_B + Q \leq 3S_m$) at a specific location, the structural action may be calculated on a plastic basis and the design will be considered to be acceptable if shakedown occurs, as opposed to continuing deformation, and if the deformations which occur prior to shakedown do not exceed specified limits, as per paragraph N417.6(a) (2) of the ASME B&PV Code, Section III, Nuclear Vessels - 1968 Edition.

Note 3: The limits on local membrane stress intensity ($P_L \leq 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) + PB $\leq 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings do not exceed 120 percent of 2/3 of the lower bound collapse load as per paragraph N417.10(c) of the ASME B&PV Code, Section III, Nuclear Vessels - 1968 Edition.

TABLE 2.9-2 (Cont'd)

NOTES

NOTE 4: A plastic instability analysis may be performed for specific cases considering the actual strain-hardening characteristics of the material, but with yield strentgh adjusted to correspond to the tabulated value at the appropriate temperature in Table N-424 or N-425, as per paragraph N-417.11(c) of the ASME B&PV Code, Section III, Nuclear Vessel - 1968 Edition.

3.0 REACTOR

3.1 SUMMARY DESCRIPTION

The Cycle 13 reactor core contains three regions of fuel in a low leakage loading pattern as described in Section 3.5.2. The fuel rods are cold worked, partially annealed Zircaloy tubes containing slightly enriched uranium dioxide fuel.

All fuel rods are pressurized with helium during fabrication to reduce stresses and strains and to increase fatigue life.

The fuel assembly is a canless type with the basic assembly consisting of the RCC guide thimbles fastened to the grids, and to the top and bottom nozzles. The fuel rods are supported at several points along their length by the spring-clip grids.

Full length rod cluster control assemblies are inserted into the guide thimbles of the fuel assemblies. The absorber sections of the control rods are fabricated of silver-indium-cadmium alloy sealed in stainless steel tubes.

The control rod drive mechanisms for the full length RCC assemblies are of the magnetic latch type. The latches are controlled by three magnetic coils. They are so designed that upon a loss of power to the coils, the rod cluster control assembly is released and falls by gravity to shut down the reactor.

The reactor was initially supplied with fuel from Westinghouse Electric Corp. (W). Reload fuel for Cycles 2 through 7 was supplied by Exxon Nuclear Co (ENC). Cycles 8 through 13 reload fuel was supplied by Westinghouse Electric Corp. The latest information regarding the current fuel cycle may be found in Sub-Chapter 3.5.

In addition to this summary description, this chapter contains: a description of the mechanical components of the reactor and reactor core, including Cycle 1 W fuel assemblies, reactor internals and control rod mechanisms (Sub-Chapter 3.2); a description of the Cycle 1 nuclear design for the W fuel (Sub-Chapter 3.3); a description of the Cycle 1 thermohydraulic design (Sub-Chapter 3.4); and a description of the current core design (Sub-Chapter 3.5).

The information contained in this chapter is principally concerned with the nuclear fuel and reactor internals design and therefore does not necessarily reflect the same information as that used in the safety analysis. For information concerning safety analysis, Chapter 14 should be consulted.

3.1.1 Performance Objectives

The current licensed thermal power limit is 3250 MWt. Calculations indicate that hot channel factors are considerably less than those used for design purposes in this application. The thermal and hydraulic design, and accident analyses (except large break LOCA) in Chapter 14, were performed at 3411 MWt for Cycle 8. These analyses identify design/safety limits for a potential uprating.

The turbine-generator and plant heat removal systems have been designed for a thermal rating of 3391 MWt. The portions of the safety analysis dependent on heat removal capacity of plant and safeguards systems have assumed the maximum calculated power rating of 3391 MWt, as have the evaluations of activity release and radiation exposure.

The initial reactor core fuel loading was designed to yield the first cycle nominal burnup of 16,666 MWD/MTU, and the Cycle 2 through 7 reload designs yield an nominal cycle burnup of 10,000 MWD/MTU. Reload designs

for Cycles 8 through 13, yield nominal cycle burnups of between 15,000 and 16,000 MWD/MTU. The fuel rod cladding is designed to maintain its integrity for the anticipated core life. The effects of gas release, fuel dimensional changes, and corrosion-induced or irradiation-induced changes in the mechanical properties of cladding are considered in the design of the fuel assemblies.

Rod control clusters are employed to provide sufficient reactivity control to terminate any credible power transient prior to reaching the applicable design minimum departure from nucleate boiling (DNB) ratio (see Section 3.5.3). This is accomplished for the current cycle by ensuring sufficient control cluster worth to shut the reactor down by at least 1.6% in the hot condition with the most reactive control cluster stuck in the fully withdrawn position.

Redundant equipment is provided to add soluble poison to the reactor coolant in the form of boric acid to maintain shutdown margin when the reactor is cooled to ambient temperatures.

In addition, the control rod worth in conjunction with the boric acid injection from the boric acid injection tank is sufficient to prevent return to criticality as a result of the maximum credible steam break (one safety valve stuck fully open) even assuming that the most reactive control rod is in the fully withdrawn position.

Experimental measurements from critical experiments or operating reactors, or both, are used to validate the methods employed in the design. During design, nuclear parameters are calculated for various operational phases and, where applicable, are compared with design limits to show that an adequate margin of safety exists.

In the thermal hydraulic design of the core, the maximum fuel and clad temperatures during normal reactor operation and at 118% overpower have been conservatively evaluated and found to be consistent with safe operating limitations.

3.1.2 PRINCIPAL DESIGN CRITERIA

Reactor Core Design

Criterion: The reactor core with its related controls and protection systems shall be designed to function throughout its design lifetime without exceeding acceptable fuel damage limits which have been stipulated and justified. The core and related auxiliary system designs shall provide this integrity under all expected conditions of normal operation with appropriate margins for uncertainties and for specified transient situations which can be anticipated.

The reactor core, with its related control and protection systems, is designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations. This includes the effects of the loss of reactor coolant flow, trip of the turbine generator, and loss of normal feedwater and loss of all off-site power.

The reactor control and protection system is designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum departure from nucleate boiling (DNB) ratio equal to or greater than the applicable design value for the fuel.
The integrity of fuel cladding is ensured by preventing excessive fuel swelling, excessive clad heating, and excessive cladding stress and strain. This is achieved by designing the fuel rods so that the following conservative limits are not exceeded during normal operation or any anticipated transient condition:

a) Minimum DNB ratio equal to or greater than the applicable design value for the fuel. For the current cycle, the design values are given in Section 3.5.3.

b) Fuel center temperature below melting point of UO_2

c) For W fuel for the initial core and ENC reload fuel, internal gas pressure less than the nominal external pressure (2250 psia), even at the end of life. For W reload fuel in the current cycle, the rod internal gas pressure shall remain below the value which causes the fuel-cladding diametral gap to increase due to outward cladding creep during steady-state operation.

d) Clad stresses less than the Zircaloy yield strength

e) Clad strain less than 1%

f) Cumulative strain fatigue cycles less than 80% of design strain fatigue life for ENC fuel. Cumulative strain fatigue cycles are less than the design fatigue life for W reload fuel in the current cycle.

The ability of fuel designed and operated to these criteria to withstand postulated normal and abnormal service conditions is shown by analyses described in Chapter 14 to satisfy the demands of plant operation well within applicable regulatory limits.

The reactor coolant pumps provided for the plant are supplied with sufficient rotational inertia to maintain an adequate flow coastdown and prevent core damage in the event of a simultaneous loss of power to all pumps.

In the unlikely event of a turbine trip from full power without an immediate reactor trip, the subsequent reactor coolant temperature increase and volume insurge to the pressurizer results in a high pressurizer pressure trip and thereby prevents fuel damage for this transient.

A loss of external electrical load of 50% of full power or less is normally controlled by rod cluster insertion, together with a controlled steam dump to the condenser, to prevent a large temperature and pressure increase in the reactor coolant system. In this case, the overpower-overtemperature

protection would guard against any combination of pressure, temperature, and power which could result in a DNB ratio less than the applicable design value during the transient.

In neither the turbine trip nor the loss-of-flow events do the changes in coolant conditions provoke a nuclear power excursion because of the large system thermal inertia and relatively small void fraction. Protection circuits actuated directly by the coolant conditions identified with core limits are therefore effective in preventing core damage.

Suppression of Power Oscillations

Criterion: The design of the reactor core with its related controls and protection systems shall ensure that power oscillations the magnitude of which could cause damage in excess of acceptable fuel damage limits, are not possible or can be readily suppressed.

The potential for possible spatial oscillations of power distribution for this core has been reviewed. It is concluded that low frequency xenon oscillations may occur in the axial dimension, and control rods can be used to supress these oscillations. The core is expected to be stable to xenon oscillations in the X-Y dimension. Out-of-core instrumentation is provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced oscillations. (In-core instrumentation is used to periodically calibrate and verify the information provided by the out-of-core instrumentation.) The analysis, detection and control of these oscillations is discussed in Reference 2) of Sub-Chapter 3.3.

Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided.

Two independent reactivity control systems are provided, one involving rod cluster control (RCC) assemblies and the other involving chemical shimming.

Reactivity Hot Shutdown Capability

Criterion: The reactivity control systems provided shall be capable of making and holding the core subcritical from any hot standby or hot operating condition.

The reactivity control systems provided are capable of making and holding the core subcritical from any hot standby or hot operating condition, including those resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the initial core.

The rod cluster control (RCC) assemblies are divided into two categories comprising control banks and shutdown banks. The control banks used in combination with chemical shim control provide control of the reactivity changes of the core throughout the life of the core during power operation. These banks of RCC assemblies are used to compensate for short term reactivity changes at power that might be produced due to variations in reactor power level or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life, such as those due to fuel depletion and fission product buildup.

Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core subcritical under any anticipated operating condition (including anticipated operational transients) sufficiently fast enough to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure subcriticality with the most reactive control rod fully withdrawn.

The reactor core, together with the reactor control and protection system is designed so that the applicable minimum allowable DNBR value is satisified and there is no fuel melting during normal operation, including anticipated transients.

The shutdown groups are provided to supplement the control groups of RCC assemblies to make the core at least 1.6 percent subcritical at the hot zero power condition (k_{eff} = 0.984) following trip from any credible operating condition, assuming the most reactive RCC assembly is in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core subcritical, assuming the most reactive rod to be in the fully withdrawn position for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass, or relief valve, or safety valve stuck open. This is achieved by the combination of control rods and automatic boric acid addition via the emergency core cooling system. The design minimum shutdown margin is 1.6 percent, assuming the maximum worth control rod is in the fully withdrawn position, and allowing 10% uncertainty in the control rod worth calculations.

Manually controlled boric acid addition is used to maintain the shutdown margin for the long term conditions of xenon decay and plant cooldown. Redundant equipment is provided to guarantee the capability of adding boric acid to the reactor coolant system.

Reactivity Holddown Capability

Criterion: The reactivity control systems provided shall be capable of making the core subcritical under credible accident conditions with appropriate margins for contingencies, and shall be capable of limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public.

Currently, normal reactivity shutdown capability is provided within 2.4 seconds following a trip signal by control rods, with boric acid injection used for the long term xenon decay transient and for plant cooldown. As discussed in response to the previous criteria, the shutdown capability prevents return to critical as a result of the cooldown associated with a safety valve stuck fully open.

Any time that the reactor is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay. Boric acid is pumped from the boric acid tanks by one of two boric acid transfer pumps to the suction of one of three charging pumps which inject boric acid into the reactor coolant. Any charging pump and either boric acid transfer pump can be operated from diesel generator power on loss of station power. Boric acid can be injected by one pump at a rate which takes the plant to 1% shutdown in the hot condition with no rods inserted in less than 90 minutes. Enough boric acid can be injected to compensate for xenon decay although xenon decay below the equilibrium operating level does not begin until approximately 15 hours after shutdown. If two boric acid pumps are available, these time periods are halved. Additional boric acid injection is employed if it is desired to bring the reactor to cold shutdown conditions.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability, independent of control rod clusters which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components, thus achieving the measure of reliability implied by the criterion.

Alternately, boric acid solution at lower concentration can be supplied from the refueling water storage tank. This solution can be transferred directly by the charging pumps or alternately by the safety injection pumps.

The reduced boric acid concentration lengthens the time required to achieve equivalent shutdown.

Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

The reactor protection systems are capable of protecting against any single credible malfunction of the reactivity control system, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits.

Reactor shutdown with rods is completely independent of the normal rod control functions since the trip breakers completely interrupt the power to the rod mechanisms regardless of existing control signals.

Details of the effects of continuous withdrawal of a control rod and continuous deboration are described in Chapters 14 and 9 respectively.

Maximum Reactivity Worth of Control Rods

Criterion: Limits, which include reasonable margin, shall be placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals sufficiently to lose capability of cooling the core.

Limits, which include considerable margin, are placed on the maximum reactivity worth of control rods or elements and on rates at which reactivity can be increased to ensure that the potential effects of a sudden or large change of reactivity cannot (a) rupture the reactor coolant pressure boundary or (b) disrupt the core, its support structures, or other vessel internals so as to lose capability to cool the core.

The reactor control system employs control rod clusters. A portion of these are designated shutdown rods and are fully withdrawn during power operation. The remaining rods comprise the control groups which are used to control reactivity changes due to load changes and to control reactor coolant temperature. The rod cluster drive mechanisms are wired into preselected groups, and are therefore prevented from being withdrawn in other than their respective groups. The rod drive mechanism is of the magnetic latch type and the coil actuation is sequenced to provide variable speed rod travel. The maximum reactivity insertion rate is

analyzed in the detailed plant analysis assuming two of the highest worth groups to be accidentally withdrawn at maximum speed, yielding reactivity insertion rates of the order of 7.5×10^{-4} $\Delta k/k$/sec, which is well within the capability of the overpower-overtemperature protection circuits to prevent core damage.

No single credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 72 steps per minute (~45 inches per minute).

3.1.3 SAFETY LIMITS

The reactor is capable of meeting the performance objective throughout core life under both steady state and transient conditions without violating the integrity of the fuel elements. Thus the release of unacceptable amounts of fission products to the coolant is prevented.

The limiting conditions for operation established in the Technical Specifications specify the functional capacity of performance levels permitted to assure safe operation of the facility.

Design parameters which are pertinent to safety limits are specified below for the nuclear, control, thermal and hydraulic, and mechanical aspects of the design.

Nuclear Limits

At full power, the current predicted nuclear heat flux hot channel factor, F_Q does not exceed 2.15 for W fuel. The equations and curves which show the F_Q limits as a function of power and fuel height are defined in the Core Operating Limits Report and in Section 3.2.2 of the Cook Nuclear Plant Unit 1 Technical Specifications.

For any condition of power level, coolant temperature, and pressure which is permitted by the control and protection system during normal operation and anticipated transients, the hot channel power distribution is such that the

minimum DNB ratio is greater than or equal to the applicable design value given in Section 3.5.3.

Reactivity Control Limits

The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are met:

a. A minimum hot shutdown margin as shown in the Technical Specifications is available assuming a 10% uncertainty in the control rod calculation.

b. This shutdown margin is maintained with the most reactive RCCA in the fully withdrawn position.

c. The shutdown margin is maintained at ambient temperature by the use of soluble poison.

Thermal and Hydraulic Limits

The reactor core is designed to meet the following limiting thermal and hydraulic criteria:

a. The minimum allowable DNBR during normal operation, including anticipated transients, is not less than the applicable DNBR design limit. For the current cycle, design limit is given in Section 3.5.3.

b. No fuel melting during any anticipated operating condition.

To maintain fuel rod integrity and prevent fission product release, it is necessary to prevent clad overheating under all operating conditions. This is accomplished by preventing a departure from nucleate boiling (DNB) which causes a large decrease in the heat transfer coefficient between the fuel rods and the reactor coolant resulting in high clad temperatures.

The ratio of the heat flux causing DNB at a particular core location, as predicted by the W-3 and WRB-1 correlations, to the existing heat flux at the same core location is the DNB ratio. The applicable design limit DNB ratio for W and ENC fuel corresponds to a 95% probability at a 95% confidence level that DNB does not occur and is chosen to maintain an appropriate margin to DNB for all operating conditions.

Mechanical Limits

Reactor Internals

The reactor internal components are designed to withstand the stresses resulting from startup, steady state operation with any number of pumps running, and shutdown conditions. No damage to the reactor internals occurs as a result of loss of pumping power.

Lateral deflection and torsional rotation of the lower end of the core barrel is limited to prevent excessive movements resulting from seismic disturbances and thus prevent interference with rod control cluster assemblies. Core drop in the event of failure of the normal supports is limited so that the rod cluster control assemblies do not disengage from the fuel assembly guide thimbles.

The internals are further designed to maintain their functional integrity in the event of a major loss-of-coolant accident. The dynamic loading resulting from the pressure oscillations because of a loss-of-coolant accident does not cause sufficient deformation to prevent rod cluster control assembly insertion.

Fuel Assemblies

The fuel assemblies are designed to perform satisfactorily throughout their lifetime. The loads, stresses, and strains resulting from the combined effects of flow induced vibrations, earthquakes, reactor pressure, fission gas pressure, fuel growth, thermal strain, and differential expansion during both steady state and transient reactor operating

conditions have been considered in the design of the fuel rods and fuel assembly. The assembly is also structurally designed to withstand handling and shipping loads prior to irradiation, and to maintain sufficient integrity at the completion of design burnup to permit safe removal from the core, subsequent handling during cooldown, shipment and fuel reprocessing.

The fuel rods are supported at seven locations along their length within the fuel assemblies by grid assemblies which are designed to maintain control of the lateral spacing between the rods throughout the design life of the assemblies. The magnitude of the support loads provided by the grids are established to minimize possible fretting without overstressing the cladding at the points of contact between the grids and fuel rods and without imposing restraints of sufficient magnitude to result in buckling or distortion of the rods.

The fuel rod cladding is designed to withstand operating pressure loads without rupture and to maintain encapsulation of the fuel throughout the design life.

<u>Rod Cluster Control Assemblies</u>

The criteria used for the design of the cladding on the individual absorber rods in the rod cluster control assemblies (RCCA) are similar to those used for the fuel rod cladding. The cladding is designed to be free standing under all operating conditions and will maintain encapsulation of the absorber material throughout the absorber rod design life. Allowance for wear during operation is included for the RCCA cladding thickness.

Adequate clearance is provided between the absorber rods and the guide thimbles which position the rods within the fuel assemblies so that coolant flow along the length of the absorber rods is sufficient to remove the heat generated without overheating of the absorber cladding.

The clearance is also sufficient to compensate for any misalignment between the absorber rods and guide thimbles and to prevent mechanical interference between the rods and guide thimbles under any operating conditions.

Control Rod Drive Assembly

Each control rod drive assembly is designed as a hermetically sealed unit to prevent leakage of reactor coolant. All pressure-containing components are designed to meet the requirements of the ASME Code, Section III, Nuclear Vessels for Class A vessels.

The control rod drive assemblies for the full length rods provide rod cluster control assembly insertion and withdrawal rates consistent with the required reactivity changes for reactor operational load changes. This rate is based on the worths of the various rod groups, which are established to limit power-peaking flux patterns to design values. The maximum reactivity addition rate is specified to limit the magnitude of a possible nuclear excursion resulting from a control system or operator malfunction. Also, the control rod drive assemblies for the full length rods provide a fast insertion rate during a "trip" of the RCC assemblies which results in a rapid shutdown of the reactor for conditions that cannot be handled by the reactor control system.

Reactor Internals

Design Description

The reactor internals are designed to support and orient the reactor core fuel assemblies and control rod assemblies, absorb the control rod dynamic loads and transmit these and other loads to the reactor vessel flange, provide a passageway for the reactor coolant, and support in-core instrumentation. The reactor internals are shown in Figure 3.2.1-2.

The internals are designed to withstand the forces due to weight, preload of fuel assemblies, control rod dynamic loading, vibration, and earthquake acceleration. These internals are analyzed in a manner similar to Connecticut Yankee, San Onofre, Zorita, Saxton and Yankee. Under the loading conditions, including conservative effects of design earthquake loading, the structure satisfies stress values prescribed in Section III, ASME Nuclear Vessel Code.

The reactor internals are equipped with bottom-mounted in-core instrumentation supports. These supports are designed to sustain the applicable loads outlined above.

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure and the in-core instrumentation support structure.

Lower Core Support Structure

The major containment and support member of the reactor internals is the lower core support structure, shown in Figure 3.2.1-5. This support structure assembly consists of the core barrel, the core baffle, and lower core plate and support columns, the thermal shield,

the intermediate diffuser plate and the bottom support plate which is welded to the core barrel. All the major material for this structure is Type 304 Stainless Steel. The core support structure is supported at its upper flange from a ledge in the reactor vessel head flange and its lower end is restrained in its transverse movement by a radial support system attached to the vessel wall. Within the core barrel are axial baffle and former plates which are attached to the core barrel wall and form the enclosure periphery of the assembled core. The lower core plate is positioned at the bottom level of the core below the baffle plates and provides support and orientation for the fuel assemblies.

The lower core plate is a 2" inch thick member through which the necessary flow distributor holes for each fuel assembly are machined. Fuel assembly locating pins (two for each assembly) are also inserted into this plate. Columns are placed between this plate and the bottom support plate of the core barrel in order to provide stiffness and to transmit the core load to the bottom support plate. Intermediate between the support plate and lower core support plate is positioned a perforated plate to diffuse uniformly the coolant flowing into the core.

The one piece thermal shield is fixed to the core barrel at the top with rigid bolted connections. The bottom of the thermal shield is connected to the core barrel by means of axial flexures. This bottom support allows for differential axial growth of the shield/core barrel but restricts radial or horizontal movement of the bottom of the shield. Rectangular tubing in which material samples can be inserted and irradiated during reactor operation are welded to the thermal shield and extend to the top of the thermal shield. These samples are held in the rectangular tubing by a preloaded spring device at the top and bottom.

Substantial scale model testing was performed at Westinghouse. This included tests which involved a complete full scale fuel assembly which was operated at reactor flow, temperature and pressure conditions. Tests were run on a 1/7th scale model of the Indian Point Unit 2 reactor. Measurements taken from these tests indicate very little shield movement, on the order of a few mils when scaled up to Indian Point Unit 2. Strain gauge measurements taken on the core barrel also indicate very low stresses. Testing to determine thermal shield excitation due to inlet flow disturbances have been included. Information gathered from these tests was used in the design of the thermal shield and core barrel.

In order to provide further confirmation of the internals design, Indian Point Unit 2 had deflection gauges mounted on the thermal shield top and bottom for the hot-functional tests. Six such gauges were mounted in the top of the thermal shield equidistant between the fixed supports and eight located at the bottom, equidistant between the six flexures, and two next to flexure supports. The internals inspection, just before the hot-functional tests, included looking at mating bearing surfaces, main welds and welds used on bolt locking devices. At the conclusion of the hot-functional tests, measurement readings were taken from the deflectometers on the shield and the internals were re-examined at all key areas for any evidence of malfunction. It can be concluded from the testing programs, analyses and the experience gained from Indian Point Unit 2, that the design as employed on this plant is adequate.

Core Components

Core components for the initial core are discribed in the following subsections. The current Westinghouse Company reload fuel is described in Section 3.5.

Design Description

Westinghouse Fuel Assembly

All of the Westinghouse fuel assemblies which have been in the core were of similar design. The overall configuration of the fuel assemblies is shown in Figures 3.2.1-8 and 3.2.1-9. The assemblies are square in cross-section, nominally 8.426 inches on a side, and have an overall height of 160.1 inches.

The fuel rods in a fuel assembly are arranged in a square array with 15 rod locations per side and a nominal centerline-to-centerline pitch of 0.563 inch between rods. Of the total possible 225 rod locations per assembly, 20 are occupied by guide thimbles for the RCCA rods and one for in-core instrumentation. The remaining 204 locations contain fuel rods. In addition to fuel rods, a fuel assembly is composed of a top nozzle, a bottom nozzle, 7 grid assemblies, 20 absorber rod guide thimbles, and one instrumentation thimble.

The guide thimbles in conjunction with the grid assemblies and the top and bottom nozzles comprise the basic structural fuel assembly skeleton. The top and bottom ends of the guide thimbles are secured to the top and bottom nozzles respectively. The grid assemblies, in turn, are fastened to the guide thimbles at each location along the height of the fuel assembly at which lateral support for the fuel rods is required. Within this skeletal framework the fuel rods are contained and supported and the rod-to-rod centerline spacing is maintained along the assembly.

Bottom Nozzle

The bottom nozzle is a square box-like structure which controls the coolant flow distribution to the fuel assembly and functions as the bottom structural element of the fuel assembly. The nozzle, which is square in cross-section, is fabricated from Type 304 stainless steel

The complete drive mechanism, shown in Figure 3.2.1-13, consists of the internal (latch) assembly, the pressure vessel, the operating coil stack, the drive shaft assembly, and the rod position indicator coil stack.

Each assembly is an independent unit which can be dismantled or assembled separately. Each mechanism pressure housing is threaded onto an adaptor on top of the reactor pressure vessel and seal welded*. The operating drive assembly is connected to the control rod (directly below) by means of a grooved drive shaft. The upper section of the drive shaft is suspended from the working components of the drive mechanism. The drive shaft and control rod remain connected during reactor operation, including tripping of the rods.

Main coolant fills the pressure containing parts of the drive mechanism. All working components and the shaft are immersed in the main coolant.

Three magnetic coils, which form a removable electrical unit and surround the rod drive pressure housing induce magnetic flux through the housing wall to operate the working components. They move two sets of latches which lift, lower and hold the grooved drive shaft.

The three magnets are turned on and off in a fixed sequence by solid-state switches for the full length rod assemblies.

The sequencing of the magnets produces step motion over the 144 inches of normal control rod travel.

The mechanism develops a lifting force approximately two times the static lifting load. Therefore, extra lift capacity is available for overcoming mechanical friction between the moving and the stationary parts. Gravity provides the drive force for rod insertion and the weight of the whole rod assembly is available to overcome any resistance.

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*A leak in a CRDM lower canopy seal weld was repaired using a mechanical seal clamp.

The mechanisms are designed to operate in water at 650°F and 2485 psig. The temperature at the mechanism head adaptor will be much less than 650°F because it is located in a region where there is limited flow of water from the reactor core, while the pressure is the same as in the reactor pressure vessel.

A multi-conductor cable connects the mechanism's operating coils to the 125 volt d-c power supply.

Latch Assembly

The latch assembly contains the working components which withdraw and insert the drive shaft and attached control rod. It is located within the pressure housing and consists of the pole pieces for three electromagnets. They actuate two sets of latches which engage the grooved section of the drive shaft.

The upper set of latches move up or down to raise or lower the drive rod by 5/8 inch. The lower set of latches have a maximum 1/16 inch axial movement to shift the weight of the control rod from the upper to the lower latches.

Pressure Vessel

The pressure vessel consists of the pressure housing and rod travel housing. The pressure housing is the lower portion of the vessel and contains the latch assembly. The rod travel housing is the upper portion of the vessel. It provides space for the drive shaft during its upward movement as the control rod is withdrawn from the core.

e) Pelleṫ-to-pellet gaps

All fuel rods are inspected by gamma scanning or other approved methods to ensure that no significant gaps exist between pellets.

f) Gamma Scanning

All fuel rods are active gamma scanned to verify enrichment control prior to acceptance for assembly loading.

g) Traceability

Traceability of rods and associated rod components is established by Quality Control.

4) Rod Upgrading

The rods, upon final inspection, are upgraded and available for fuel assembly loading.

5) Assembly

Inspection consists of 100 percent inspection of drawing requirements.

6) Other Inspection

The following inspection is performed as part of routine inspection operation:

a) Measurements other than those specified above which are critical to thermal and hydraulic analyses are obtained to enable evaluation of manufacturing variations to a 99.5% confidence level.

b) Tool and gauge inspection and control including standardization to primary and secondary working standards. Tool inspection is performed at prescribed intervals on all serialized tools. Complete records are kept of calibration and condition of tools.

c) Check audit inspection of all inspection activities and records to assure that prescribed methods are followed and that all records are correct and properly maintained.

d) Surveillance of outside contractors, including approval of standards and methods is performed where necessary. However, all final acceptance is based upon inspection performed at the Westinghouse plant.

To prevent the possibility of mixing enrichments during fuel manufacture and assembly, meticulous process control is exercised.

The UF6 is received from the DOE diffusion plant in 5000 lb cylinders. These cylinders are tagged with the enrichment of the contents. In addition, samples of the contents are attached. These samples are analyzed by Westinghouse to verify the enrichment of the contents.

Following verifications, the cylinders are moved to the production area, where they are piped in to the UF6 to U02 conversion process equipment and thereafter (during the conversion of the particular region of the core) remain a permanent part of the process equipment. Upon completion of this conversion, the U02 is placed into sealed containers which are color coded to identify the enrichment of the contents.

Movement of powder from the conversion area to the pellet production area can be made by one authorized group only who direct the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single enrichment and density are produced in a given production line.

Finished pellets are placed on trays having the same color code as the powder containers and transferred to segregated storage racks. Physical barriers prevent mixing of pellets of different densities and enrichments in this storage area. Unused powder and substandard pellets to be reprocessed are returned to storage in the original color coded containers.

Loading of the pellets into the cladding is again accomplished in isolated production lines and again only one density and enrichment is loaded on a line at a time.

At the time of loading, the top fuel tube end plug identification character is checked with the density and enrichment identification of the color code of the pellet storage tray. After each fuel tube is seal welded, it is given the same color coding as has been carried throughout the previous processes. The fuel tube remains color coded until just prior to installation in the fuel assembly. The color coding and end plug identification character provide a cross reference of the fuel contained in the fuel rods.

At the time of installation into an assembly, the color coding is removed. After the fuel rods are installed, an inspector verifies that all fuel rods in an assembly have the same end plug identification, and that the top nozzle to be used on the assembly carries the correct identification character describing the fuel enrichment and density for the core region being fabricated. The top nozzle identification then becomes the permanent description of the fuel contained in the assembly.

<u>Burnable Poison Rod Tests and Inspections</u>

The end plug seal welds are checked for integrity by visual inspection and X-ray. The finished rods are helium leak checked.

REFERENCES, SECTION 3.2.1

1. Daniel, R. C., et al, "Effects of High Burnup on Zircaloy-Clad Bulk UO2, Plate Fuel Element Samples, "WAPD-263, (September, 1965).

2. Large Closed Cycle Water Reactor Research and Development Program Quarterly Progress Reports for the Period January 1963 through June 1965 (WCAP-3738, 3739, 3743, 3750, 3269-2, 3269-3, 3269-5, 3269-6, 3269-12 and 3269-13).

3. J. S. Moore, WCAP-9000 "Nuclear Design of Westinghouse PWR's with Burnable Poison Rods", March 1969.

4. WCAP-7072 "Use of Part Length Absorber Rods in Westinghouse Pressurized Water Reactors".

TABLE 3.3.1-1 (cont'd.)

BURNABLE POISON RODS

39.	Number and Material	1436 Borosilicate Glass
40.	Worth Hot Full Power $\Delta\rho$	9.0%
41.	Worth Cold $\Delta\rho$	7.0%

KINETIC CHARACTERISTICS

42.	Moderator Temperature Coefficient at Full Power ($\Delta\rho/{}^\circ F$)	$-\ .3 \times 10^{-4}$ to $-\ 3.2 \times 10^{-4}$
43.	Moderator Pressure Coefficient ($\Delta\rho$/psi)	$+\ .3 \times 10^{-6}$ to 4.0×10^{-6}
44.	Moderator Density Coefficient, $\Delta\rho/gm/cm^3$	$-\ .1 \times 10^{-5}$ to 0.8×10^{5}
45.	Doppler Coefficient ($\Delta\rho/{}^\circ F$)	$-\ 1.0 \times 10^{-5}$ to $-\ 1.7 \times 10^{-5}$
46.	Delayed Neutron Fraction, %	.51 to .70
47.	Prompt Neutron Lifetime, sec.	1.4×10^{-5} to 2.0×10^{-5}
48.	Boron Worth $\Delta\rho$/ppm	1.4×10^{-4} $.09 \times 10^{-4}$

TABLE 3.3.1-2

REACTIVITY REQUIREMENTS FOR CONTROL RODS

Requirements	Per Cent $\Delta\rho$ Beginning of Life	End of Life
Control		
Power Defect	1.70	3.05
Rod Insertion Limit	0.70	0.50
Total Control	2.40	3.55

TABLE 3.5.1-1

Westinghouse 15x15 OFA Design Parameters

Parameter	REF. 15x15 W Optimized Fuel Assembly Design	Region 14
Fuel Assembly Length, in.	159.785	159.975
Fuel Rod Length, in.	151.85	152.17
Assembly Envelope, in.	8.426	
Compatible with Core Internals	Yes	
Fuel Rod Pitch, in.	0.563	
Number of Fuel Rods/Ass'y	204	
Number of Guide Thimbles/Ass'y	20	
Number of Instrumentation Tube Ass'y	1	
Compatible with Moveable In-Core	Yes	
Detector System Fuel Tube Material	Zircaloy-4	
Fuel Rod Clad OD, in.	0.422	
Fuel Rod Clad Thickness, in.	0.0243	
Fuel/Clad Gap, mil	7.5	
Fuel Pellet Diameter, in.	0.3659	
Guide Thimble Material	Zircaloy-4	
Guide Thimble ID, in.*	0.499	
Structural Material - Five Inner Grids	Zircaloy-4	
Structural Material - Two End Grids	Inconel	
Grid height, in., Outer Straps, Valley-to-Valley	2.25 (Inner Grids) 1.50 (End Grids)	
Bottom Nozzle	Reconstitutable	
Top Nozzle Holddown Springs	3-leaf	

*Above dashpot

Table 3.5.1-2 intentionally deleted.

3.5.2 NUCLEAR DESIGN

The nuclear design of cores with W OFA is accomplished by using the standard calculational methods as described in the W Reload Safety Evaluation Methodology. In addition to Westinghouse's standard methods, the Westinghouse Advanced Nodal Code (ANC)[9] was introduced in Cycle 11 to perform core neutronics analyses and the PHOENIX-P code[3] was introduced in Cycle 12 to calculate lattice physics constants.

Each reload core design is evaluated to assure that design and safety limits for the fuel are satisfied according to the W reload safety evaluation methodology. For the evaluation of the worst-case $F_Q(Z)$ envelope, axial power shapes are synthesized with the limiting F_{xy} values chosen over three overlapping burnup windows during the cycle.

In order to accommodate potential increases in future feed enrichments, a criticality analysis of the fuel storage areas was performed for nominal enrichments in W 15x15 OFA fuel up to and including 4.55 wt.% U-235 for the new fuel storage vault and 4.95 wt.% U-235 for the spent fuel pool. These analyses confirm that all current safety criteria applicable to fuel storage are satisfied.[2]

3.5.2.1 Computerized Methods, Codes and Cross Section Data

Three principal computer codes have been used in the nuclear design of reactor cores with W OFA; these are PHOENIX-P (two-dimensional), APOLLO (one-dimensional), and ANC (two-dimensional and three-dimensional). Descriptions and uses for these codes follow.

PHOENIX-P[3] is a two-dimensional multi-group transport theory code used to calculate lattice physics constants. Microscopic cross section data are based on a 42-energy group structure that has been derived from the CSRL-V 227 group ENDF/B-V library[4]. It provides the capability for cell lattice modeling on an assembly level. In the core design, PHOENIX-P is used to provide homogenized, two-group cross-sections for nodal calculations and feedback models. It is also used in a special geometry to generate appropriately weighted constants for the baffle/reflector regions.

APOLLO, an advanced version of PANDA[5], is a two-group, one-dimensional diffusion-depletion code. APOLLO utilizes the burnup dependent radially averaged macroscopic cross sections of the corresponding 3-D model. The APOLLO model is used as an axial model. APOLLO is utilized to determine axial power and burnup distributions, differential rod worths, and control rod operational limits (insertion limits, return to power limits, etc.).

ANC[6] is an advanced nodal code that is used in two-dimensional and three-dimensional calculations. ANC calculations include power and burnup distributions, critical boron concentrations, reactivity coefficients, control rod worths, and various safety analysis calculations. ANC is used to validate one dimensional results from APOLLO and to provide information about radial (X-Y) peaking factors as a function of axial position. ANC also has the capability of calculating discrete pin powers and pin burnups from the nodal information.

Additional support codes are used for special calculations such as determining fuel temperatures.

3.5.2.2 Neutronic Design of Cook Nuclear Plant Unit 1 Reactor Core

3.5.2.2.1 Analytical Input

The neutronics design methods utilized to calculate the data presented herein are consistent with those described previously with primary reliance upon the ANC code.

For each cycle, the burnup history of each of the fuel assemblies retained from previous cycles for further energy production is calculated by a three-

dimensional model which is utilized to simulate operation of the core for previous cycles.

As an example, Cycle 13 core calculations used assembly exposures calculated from the Cycle 12 burnup of 16,541 MWD/MTU.

3.5.2.2.2 Design Bases

For each cycle, the nuclear design bases are very similar to those for the example Cycle 13 core as follows:

1. At core full power, 3250 MWt (not including pump heat), nuclear peaking factors of 2.15 and 1.55 for F_Q^T and $F_{\Delta H}^N$ respectively, will not be exceeded. In addition, at any relative power level P $(0.0 \leq P \leq 1.0)$, F_Q^T and $F_{\Delta H}^N$ shall not exceed the bases of the plant control and protection system.

2. The moderator temperature coefficient at operating conditions greater than 70% power level is a ramp function limited to +5.0 pcm/$^{\circ}$F at 70% power and 0.0 pcm/$^{\circ}$F at 100% power. Below 70% power level, the moderator temperature coefficient shall be less than +5.0 pcm/$^{\circ}$F.

3. With the most reactive control rod stuck out of the core, the remaining control rods shall be able to shut the reactor down by a sufficient reactivity to reduce the consequences of any credible accident to acceptable levels.

4. The effects of all accident situations in Cycle 13 will be acceptable and compatible with the safety bases of the Final Safety Analysis Report (FSAR), as specified in Reference 7.

5. The fuel loading specified shall be capable of generating approximately 15360 MWD/MTU at normal full power operating conditions during Cycle 13.

3.5.2.2.3 Design Description and Results

Each cycle's reactor core consists of 193 W OFA assemblies, each having a 15x15 fuel rod array. A description of the W OFAs is given in Section 3.5.1.

As an example, the Cycle 13 loading pattern is given in Figure 3.5.2-1 which shows the region number, sources, and the burnable absorber configuration. The core consists of 48 fresh W OFAs with an average enrichment of 3.099 w/o U-235, 32 fresh OFAs with an average enrichment of 3.609 w/o, 80 once burnt OFA assemblies, and 33 twice burnt OFA assemblies. A low leakage loading pattern was developed which results in the scatter-loading of the fresh OFAs throughout the interior of the core. 3328 new IFBA rods are present in a number of OFAs to control power peaking and MTC. The IFBA rods contain approximately 0.0018 gm/in of B-10. Pertinent fuel assembly parameters for the Cycle 13 fuel are given in Tables 3.5.1-1 and 3.5.2-1.

Physics Characteristics

The neutronics characteristics of a reactor core with W OFA fuel are presented in Table 3.5.2-2. These reactivity coefficients are bounded by the coefficients used in the safety analysis. For an example cycle length, Cycle 12 was projected to be 15,360 MWD/MTU at a core power of 3,250 MWt with ~ 10 ppm soluble boron remaining.

Power Distribution Considerations

Figure 3.5.2-2 shows the K(Z) function (fuel height limit for normalized $F_Q(Z)$). Each cycle's core loading satisfies the envelope shown in Figure 3.5.2-2.

Control Rod Reactivity Requirements

The Cook Nuclear Plant Unit 1 Technical Specifications require a minimum shutdown margin of 1,600 pcm in operational Modes 1, 2, 3 and 4 and 1000 pcm in operational Mode 5 at BOC and EOC. As an example, detailed calculations of shutdown margins for Cycle 12 are presented in Table 3.5.2-3. The Cycle 12 analysis indicates excess shutdown margin of 1817 pcm at BOC and 1874 pcm at EOC.

Insertion limits are specified for the control rod groups and are given in the Core Operating Limits Report, as described in Technical Specification 6.9.1.11. The control rod shutdown requirements allow for a HFP D-Bank insertion equivalent to 500 pcm at both BOC and EOC. Table 3.5.2-3 gives the shutdown requirements for the example of Cycle 12.

Moderator Temperature Coefficient

Core loadings must satisfy the Technical Specifications requirements that the moderator temperature coefficient be less than or equal to +5 pcm/$^{\circ}$F below 70% of rated thermal power and less than or equal to a linear ramp between +5 pcm/$^{\circ}$F at 70% power and 0 pcm/$^{\circ}$F at 100% power.

REFERENCES, SECTION 3.5.2

1. Bordelon, F. M., et al., "Westinghouse Reload Safety Evaluation Methodology," WCAP-9272 (Prop.) and WCAP-9273 (Non-Prop.), March 1978.

2. Alexich, M. P. to Murley, T. E., "Proposed Units 1 and 2 License Conditions and Technical Specifications Changes for Unit 2 Cycle 8 Spent Fuel Pool and New Fuel Storage Vault," AEP:NRC:1071F, December 8, 1989.

3. Nguyen, T. Q., et al. "Qualification of the PHOENIX-P/ANC Nuclear Design System for Pressurized Water Reactor Cores, WCAP-11596, November 1987.

4. Ford, W. E., et al., "CSRL-V: Processed ENDF/B-V 227-Neutron Group and Point-wise Cross Section Libraries for Criticality Safety, Reactor and Shielding Studies, NUREG/CR-2306, ORNL/CSD/TM-160, (1982).

5. R. F. Barry, C. C. Emery, and T. D. Knight, "The PANDA Code," WCAP-7048, (April 1967).

6. Y. S. Liu, A. Meliksetion, J. A. Rathkopf, D. C. Little, F. Nakano, and M. J. Poploski, "ANC - A Westinghouse Advanced Nodal Computer Code," WCAP-10965-P-A (December 1985).

7. Donald C. Cook Nuclear Plant Unit 1 Cycle 13, Reload Safety Evaluation (August 1992)

Figure 3.5.2-1: Example Core Loading Pattern:

D. C. COOK UNIT 1 CYCLE 13

180°

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					13B	14A	14B	14A	14B	14A	13B				
2			13B	15A	15B 48	15B 48 D	15A 32	15B 48 SC	15A 32	15B 48 D	15B 48	15A	13B		
3		13B	14A C	15B 48	14B A	14B	14A SB	14A SS	14A SB	14B	14B A	15B 48	14A C	13B	
4		15A	15B 48	14A SA	15A 48	13B	15A 48	14B B	15A 48	13B	15A 48	14A SA	15B 48	15A	
5	13B	15B 48	14B A	15A 48	14A	15A 48	14A	14B	14A SA	15A 48	14A	15A 48	14B A	15B 48	13B SS
6	14A	15B 48 D	14B	13B	15A 48	13A SD	15A 48	14A C	15A 48	13A SD	15A 48	13B	14B	15B 48 D	14A
7	14B	15A 32	14A SB	15A 48	14A SA	15A 48	13A	15B 48	13A	15A 48	14A	15A 48	14A SB	15A 32	14B
8	14A	15B 48 SC	14A	14B B	14B	14A C	15B 48	13B D	15B 48	14A C	14B	14B B	14A	15B 48 SC	14A
9	14B	15A 32	14A SB	15A 48	14A	15A 48	13A	15B 48	13A	15A 48	14A SA	15A 48	14A SB	15A 32	14B
10	14A	15B 48 D	14B	13B	15A 48	13A SD	15A 48	14A C	15A 48	13A SD	15A 48	13B	14B	15B 48 D	14A
11	13B SS	15B 48	14B A	15A 48	14A	15A 48	14A SA	14B	14A	15A 48	14A	15A 48	14B A	15B 48	13B
12		15A	15B 48	14A SA	15A 48	13B	15A 48	14B B	15A 48	13B	15A 48	14A SA	15B 48	15A	
13		13B	14A C	15B 48	14B A	14B	14A SB	14A SS	14A SB	14B	14B A	15B 48	14A C	13B	
14			13B	15A	15B 48	15B 48 D	15A 32	15B 48 SC	15A 32	15B 48 D	15B 48	15A	13B		
15					13B	14A	14B	14A	14B	14A	13B				

90°

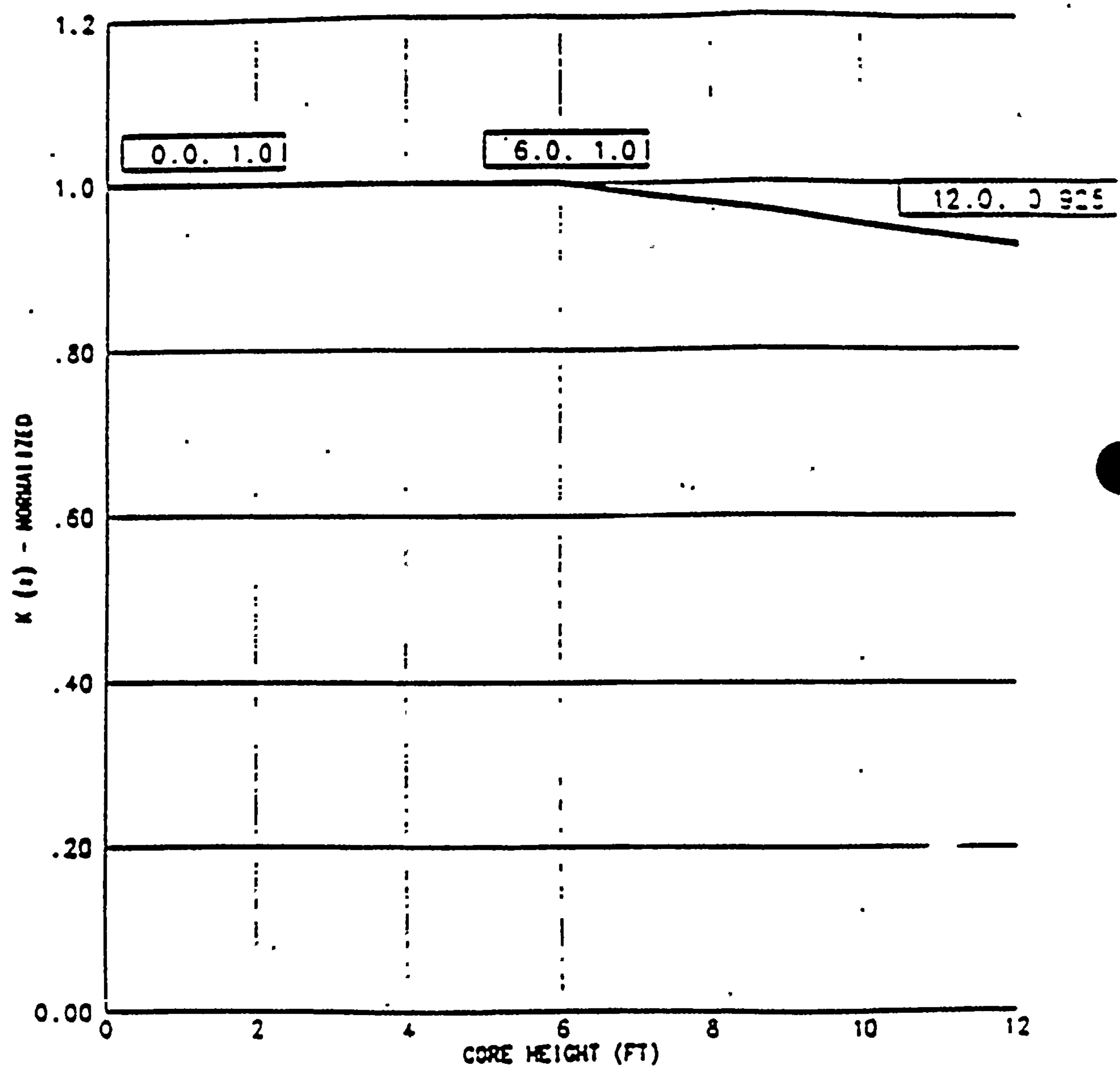
0°

Key	
X	Region Number
Y/SS	Number of IFBA, Burnable Absorbers/Secondary Sources
CTL	Control Rod Location

FIGURE 3.5.2-2

Heat Flux Hot Channel Factor
Normalized Operating Envelope, FQ ECCS Limit = 2.15

Cook Nuclear Plant Unit 1 Cycle 12



NOTES FOR TABLE 3.1-1

[a] These numbers are based on Improved Thermal Design Procedure in Reference 2.

[b] The value of 437,800 BTU/hr-ft^2 is associated with a Cycle 1 value of F_Q of 2.32. The Cycle 3 value is 375,500 BTU/hr-ft^2 corresponding to a peaking factor of 1.99.

[c] This value of 12.6 Kw/ft is associated with a Cycle 1 value of F_Q of 2.32. The Cycle 3 value is 10.98 Kw/ft associated with a peaking factor of 1.99.

[d] See Section 3.3.2.2.6.

[e] The value of F_Q = 2.32 was the value of F_Q for normal operation reported in the original FSAR. The value for Cycle 3 is 1.99.

[f] The reload feed enrichments for Cycle 8 were 1.5, 3.6 and 4.2 w/o.

[g] These numbers are based on Revised Thermal Design Procedure in Reference 3.

TABLE 3.1-2

ANALYTIC TECHNIQUES IN CORE DESIGN

Analysis	Technique	Computer Code	Section Referenced
Mechanical Design of Core Internals			
Loads, Deflections, and Stress Analysis	Static and Dynamic Modeling	Blowdown code, FORCE, Finite element structural analysis code, and others	14.3.3
Fuel Rod Design			
Fuel Performance Characteristics (temperature, internal pressure, clad stress, etc.)	Semi-empirical thermal model of fuel rod with consideration of fuel density changes, heat transfer, fission gas release, etc.	Westinghouse fuel rod design model	3.2.1.3.1 3.3.3.1 3.4.2.2 3.4.3.4.2
Nuclear Design			
1. Cross Sections and Group Constants	Microscopic data	Modified ENDF/B-V library	3.3.3.2
	Macroscopic constants for homogenized core regions	PHOENIX-P	3.3.3.2

This scheme of grid fastening is a standard for Westinghouse and has been used successfully since the introduction of Zircaloy guide thimbles in 1969.

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

Grid Assemblies

The fuel rods, as shown in Figure 3.2-2, are supported at intervals along their length by grid assemblies which maintain the lateral spacing between the rods. Each fuel rod is supported within each grid by the combination of support dimples and springs. The grid assembly consists of individual slotted straps interlocked and brazed in an "egg-crate" arrangement to join the straps permanently at their points of intersection. The straps contain spring fingers, support dimples and mixing vanes.

The grid material for the top and bottom grids is Inconel-718, chosen because of its corrosion resistance and high strength. The grid material for the mid-grids is Zircaloy, chosen because of its low neutron absorption characteristic and its extensive successful in-reactor use. The magnitude of the grid restraining force on the fuel rod is set high enough to minimize possible fretting, without overstressing the cladding at the points of contact between the grids and fuel rods. The grid assemblies also allow axial thermal expansion of the fuel rods without imposing restraint sufficient to develop buckling or distortion of the fuel rods.

The intermediate flow mixer (IFM) grids shown in Figure 3.2-2 are located in the three uppermost spans between the Zircaloy-4 mixing vane structural grids and incorporate a similar mixing vane array. The primary function of the IFM grid is to provide mid-span flow mixing in the hottest fuel assembly spans. Each IFM grid cell contains four dimples which are designed to prevent mid-span channel closure in the spans containing IFMs and fuel rod contact with

the IFM mixing vanes. This simplified cell arrangement allows for short grid cells so that the IFM grid can accomplish its flow mixing objective with minimal pressure drop.

The IFM grids are not intended to be structural members. The outer strap configuration was designed similar to current fuel designs to preclude grid hang-up and damage during fuel handling. Additionally, the grid envelope is smaller which further minimizes the potential for damage and reduces calculated forces during seismic/LOCA events. A coolable geometry is, therefore, assured at the IFM grid elevation, as well as at the structural grid elevation.

3.2.1.3 Design Evaluation

3.2.1.3.1 Fuel Rods

The fuel rods are designed to assure that the design bases are satisfied for Condition I and II events. This assures that the fuel performance and safety criteria (Section 3.2) are satisfied.

Materials - Fuel Cladding

The desired fuel rod clad is a material which has a superior combination of neutron economy (low absorption cross section), high strength (to resist deformation due to differential pressures and mechanical interaction between fuel and clad), high corrosion resistance (to coolant, fuel and fission products), and high reliability. Zircaloy-4 has this desired combination of clad properties. As shown in Reference (4), there is considerable PWR operating experience on the capability of Zircaloy as a clad material. Clad hydriding has not been a significant cause of clad perforation since current controls on levels of fuel contained moisture were instituted[4].

Materials - Fuel Pellets

Sintered, high density uranium dioxide fuel reacts only slightly with the clad at core operating temperatures and pressures. In the event of clad

The UO_2 powder is kept in sealed containers. The contents are fully identified both by descriptive tagging and preselected color coding. A Westinghouse identification tag completely describing the contents is affixed to the containers before transfer to powder storage. Isotopic content is confirmed by sample isotopic analysis.

Powder withdrawal from storage can be made by only one authorized group, which directs the powder to the correct pellet production line. All pellet production lines are physically separated from each other and pellets of only a single nominal enrichment and density are produced in a given production line.

Finished pellets are placed on trays identified with the same color code as the powder containers and transferred to segregated storage racks within the confines of the pelletizing area. Samples from each pellet lot are tested for isotopic content and impurity levels prior to acceptance by Quality Control. Physical barriers prevent mixing of pellets of different nominal densities and enrichments in this storage area. Unused powder and substandard pellets are returned to storage in the original color coded containers.

Loading of pellets into the clad is performed in isolated production lines and again only one density and enrichment is loaded on a line at a time, except when natural uranium is loaded into axial blankets. Then natural and enriched uranium pellets are separately identified by their different pellet lengths.

A serialized traceability code is placed on each fuel tube which identifies the contract and enrichment. The end plugs are inserted and inert welded to seal the tube. The fuel tube remains coded, and traceability identified until just prior to installation in the fuel assembly. The traceability code provides an identification of the fuel contained in the fuel rods.

At the time of installation into an assembly, the traceability codes are removed and a matrix is generated to identify each rod in its position within a given assembly. After the fuel rods are installed, an inspector verifies that all fuel rods in an assembly carry the correct identification character describing the fuel enrichment and density for the core region being fabricated. The top nozzle is inscribed with a permanent identification number providing traceability to the fuel contained in the assembly.

Similar traceability is provided for burnable poison, source rods and control rodlets as required.

3.2.1.4.3 Tests and Inspections by Others

If any tests and inspections are to be performed on behalf of Westinghouse, Westinghouse will review and approve the quality control procedures, inspection plans, etc. to be utilized to ensure that they are equivalent to the description provided above and are performed properly to meet all Westinghouse requirements.

3.2.1.4.4 Onsite Inspection

Onsite inspection programs for fuel, control rods and internals are based on the NSSS supplier's detailed procedures. In the event reloads or other components are supplied by other suppliers additional programs will be developed based on that supplier's procedures.

Loaded fuel containers, when received on site, are externally inspected to ensure that labels and markings are intact and seals are unbroken. After the containers are opened, the accelerometers are inspected to determine if movement during transit exceeded design limitations.

Following removal of the fuel assembly from the container in accordance with detailed procedures from the fuel fabricator, the polyethylene wrapper is removed and a visual inspection of the entire bundle is performed.

Control rod assemblies are shipped in fuel assemblies and are inspected prior to removal of the fuel assembly from the container.

Surveillance of fuel and reactor performance is routinely conducted by operating personnel. Coolant activity and chemistry are followed to permit early detection of any fuel clad defects.

Visual fuel inspection is routinely conducted during refueling. Additional fuel inspections are dependent on the results of the operational monitoring and the visual inspections.

3.2.2 REACTOR VESSEL INTERNALS

3.2.2.1 Design Bases

The design bases for the mechanical design of the reactor vessel internals components are as follows:

1. The reactor internals in conjunction with the fuel assemblies shall direct reactor coolant flow through the core to achieve acceptable flow distribution and to restrict bypass flow so that the heat transfer performance requirements are met for all modes of operation. In addition, required cooling for the pressure vessel head shall be provided so that the temperature differences between the vessel flange and head do not result in leakage from the flange during reactor operation.

2. In addition to neutron shielding provided by the reactor coolant, a separate thermal shield is provided to limit the neutron exposure of the pressure vessel material in order to maintain the required ductility of the material for all modes of operation.

3. Provisions shall be made for installing incore instrumentation useful for the plant operation and vessel material test specimens required for a pressure vessel irradiation surveillance program.

4. The reactor internals shall be designed to withstand mechanical loads arising from operating basis earthquake, safe shutdown earthquake and pipe ruptures and meet the requirement of Item 5 below.

5. The reactor shall have mechanical provisions which are sufficient to adequately support the core and internals and to assure that the core is intact with acceptable heat transfer geometry following transients arising from abnormal operating conditions.

6. Following the design basis accident, the plant shall be capable of being shutdown and cooled in an orderly fashion so that fuel cladding temperature is kept within specified limits. This implies that the deformation of certain critical reactor internals must be kept sufficiently small to prevent overstressing of fuel elements to failure plus allow adequate core cooling.

The functional limitations for the core structures during the design basis accident are shown in Table 3.2-1. To ensure no column loading of rod cluster control guide tubes, the upper core plate deflection is limited to not exceed the value shown in Table 3.2-1.

Details of the dynamic analyses, input forcing functions, and response loadings are presented in Section 14.3.3.

3.2.2.2 Description and Drawings

The reactor vessel internals are described as follows:

The components of the reactor internals are divided into three parts consisting of the lower core support structure (including the entire core barrel and thermal shield), the upper core support structure and the incore instrumentation support structure. The reactor internals support the core, maintain fuel alignment, limit fuel assembly movement, maintain alignment between fuel assemblies and control rod drive mechanisms, direct coolant flow

Part Length Rod Cluster Control Assembly

Part Length Control Rods are not installed in Unit 2 of the Donald C. Cook Nuclear Plant for the following reasons:

a. No credit is taken for their presence in the safety analysis performed by the vendor. Therefore, the decision not to mount them does not constitute a safety issue,

b. The reactor's Operating License and Technical Specifications preempt their use,

c. Unit 1 of the Donald C. Cook Nuclear Plant has successfully load followed and controlled artificially created large xenon oscillations without the use of these rods, and

d. The storage of these rods outside the reactor vessel will prevent their irradiation and subsequent radioactivity.

Anti-rotation devices will be installed in a manner similar to the practice followed during the hot functional tests. Permanent anti-rotation devices similar to those in Unit 1 will be installed later.

Burnable Absorber Assembly

Each burnable absorber assembly consists of burnable absorber rods attached to a hold down assembly. Burnable absorber assemblies are shown in Figure 3.2-15.

The absorber rods consist of borosilicate glass tubes contained within Type 304 stainless steel tubular cladding which is plugged and seal welded at the ends to encapsulate the glass. The glass is also supported along the length of its inside diameter by a thin wall tubular inner liner. The top end of the liner is open to permit the diffused helium to pass into the void volume and the liner overhangs the glass. The liner has an outward flange at the bottom end to maintain the position of the liner with the glass. A typical

burnable absorber rod is shown in longitudinal and transverse cross sections in Figure 3.2-16.

The absorber rods in each fuel assembly are grouped and attached together at the top end of the rods to a hold down assembly by a flat, perforated retaining plate which fits within the fuel assembly top nozzle and rests on the adaptor plate. The retaining plate (and the poison rods) is held down and restrained against vertical motion through a spring pack which is attached to the plate and is compressed by the upper core plate when the reactor upper internals assembly is lowered into the reactor. This arrangement ensures that the absorber rods cannot be ejected from the core by flow forces. Each rod is permanently attached to the base plate by a nut which is lock welded into place.

The clad in the rod assemblies is 10 percent cold worked Type 304 stainless steel. All other structural materials are Types 304 or 308 stainless steel except for the springs which are Inconel-718. The borosilicate glass tube provides sufficient boron content to meet the criteria discussed in Section 3.3.1.

Neutron Source Assembly

The purpose of the neutron source assembly is to provide base neutron level to ensure that the detectors are operational and responding to core multiplication neutrons. Since there is very little neutron activity during loading, refueling, shutdown and approach to criticality, a neutron source is placed in the reactor to provide a minimum neutron count of at least 2 counts per second on the source range detectors attributable to core neutrons. The detectors, called source range detectors, are used primarily when the core is subcritical and during special subcritical modes of operations.

The source assembly also permits detection of changes in the core multiplication factor during core loading, refueling and approach to criticality. This can be done since the multiplication factor is related to an inverse function of the detector count rate. Therefore, a change in the multiplication factor can be detected during addition of fuel assemblies

holding force created by a single winding is sufficient to overcome the rundown torque produced by the mechanism load. Therefore, the rod cannot move except under the control of the power supply.

The rotational energy is supplied in sequential pulses to the armature which rotates directionally 15 degrees per pulse as controlled by the power supply.

3.2.3.3 Design Evaluation

3.2.3.3.1 Reactivity Control Components

The components are analyzed for loads corresponding to normal, upset, emergency and faulted conditions. The analysis performed depends on the mode of operation under consideration.

The scope of the analysis requires many different techniques and methods, both static and dynamic.

Some of the loads that are considered on each component where applicable are as follows:

1. Control rod trip (equivalent static load).
2. Differential pressure.
3. Spring preloads.
4. Coolant flow forces (static).
5. Temperature gradients.
6. Differences in thermal expansion.
    a. Due to temperature differences.
    b. Due to expansion of different materials.
7. Interference between components.
8. Vibration (mechanically or hydraulically induced).
9. Operational transients.
10. Pump overspeed.

11. Seismic loads (operating basis earthquake and safe shutdown earthquake).
12. Blowdown forces (due to cold or hot leg break).
13. Material swelling and gas generation pressure.

The main objective of the analysis is to satisfy allowable stress limits, to assure an adequate design margin, and to establish deformation limits which are concerned primarily with the functioning of the components. The stress limits are established not only to assure that peak stresses will not reach unacceptable values, but also limit the amplitude of the oscillatory stress component in consideration of fatigue characteristics of the materials. Standard methods of strength of materials are used to establish the stresses and deflections of these components. The dynamic behavior of the reactivity control components has been studied using experimental test data and experience from operating reactors.

The design of incore component rods provides a sufficient cold void volume within the burnable absorber and source rods to limit the internal pressures to a value which satisfies the criteria in Section 3.2.3.1. The void volume for the helium in the burnable absorber rods is obtained through the use of glass in tubular form which provides a central void along the length of the rods. Helium gas is not released by the neutron absorber rod material, thus the absorber rod only sustains an external pressure during operating conditions. The internal pressure of source rods continues to increase from ambient until end of life. The stress analysis of reactivity component rods assumes 100 percent gas release to the rod void volume.

Based on available data for properties of the borosilicate glass and on nuclear and thermal calculations for the burnable absorber rods, gross swelling or cracking of the glass tubing is not expected during operation. Some minor creep of the glass at the hot spot on the inner surface of the tube could occur but would continue only until the glass came in contact with the inner liner. The wall thickness of the inner

3. All clad/end plug welds are checked for integrity by visual inspection, X-ray and are helium leak checked. All the seal welds in the neutron absorber rods, burnable absorber rods and source rods are checked in this manner.

4. To assure proper fitup with the fuel assembly, the rod cluster control, burnable absorber and source assemblies are installed in the fuel assembly without restriction or binding in the dry condition with a force not to exceed 15 pounds. Also a straightness of 0.01 in./ft. is required on the entire inserted length of each rod assembly.

   The full length rod cluster control assemblies are functionally tested, following core loading but prior to criticality to demonstrate reliable operation of the assemblies. Each assembly is operated (and tripped) one time at no flow/cold conditions and one time at full flow/hot conditions. In addition, selected assemblies, amounting to about 15 to 20 percent of the total assemblies are operated at no flow/operating temperature conditions and full flow/ambient conditions. Also the slowest rod and the fastest rod are tripped 10 times at no flow/ambient conditions and at full flow/operating temperature conditions. Thus each assembly is tested a minimum of 2 times or up to 14 times maximum to ensure the assemblies are properly functioning.

3.2.3.4.2 Control Rod Drive Mechanisms

Quality assurance procedures during production of control rod drive mechanisms include material selection, process control, mechanism component tests and inspections during production and hydrotests.

After all manufacturing procedures had been developed, several prototype control rod drive mechanisms and drive rod assemblies were life tested with the entire drive line under environmental conditions of temperature,

pressure and flow. All acceptance tests confirm the 3×10^6 step life capability of the control rod drive mechanism and drive rod assembly.

These tests include verification that the trip time achieved by the full length control rod drive mechanisms meet the design requirement of 2.7 seconds from the beginning of decay of stationary gripper goil voltage to dashpot entry. This trip time requirement will be confirmed for each control rod drive mechanism prior to initial reactor operation and at periodic intervals after initial reactor operation. In addition, a Technical Specification has been set to ensure that the trip time requirement is met.

It is expected that all control rod drive mechanisms will meet specified operating requirements for the duration of plant life with normal refurbishment. However, a Technical Specification pertaining to an inoperable rod cluster control assembly has been set.

If a rod cluster control assembly cannot be moved by its mechanism, adjustments in the boron concentration ensure that adequate shutdown margin would be achieved following a trip. Thus, inability to move one rod cluster control assembly can be tolerated. More than one inoperable rod cluster control assembly could be tolerated, but would impose additional demands on the plant operator. Therefore, the number of inoperable rod cluster control assemblies has been limited to one.

In order to demonstrate proper operation of the control rod drive mechanism and to ensure acceptable core power distributions during operation partial rod cluster control assembly movement checks are performed on the full length rod cluster control assemblies as described in the Technical Specifications. In addition, periodic drop tests of the full length rod cluster control assemblies are performed at each refueling shutdown to demonstrate continued ability to meet trip time requirements, to ensure core subcriticality after reactor trip, and to limit potential reactivity insertions from a

20. Weiner, R. A., et. al., "Improved Fuel Performance Models for Westinghouse Fuel Rod Design and Safety Evaluations," WCAP-11873-A, August 1988.

TABLE 3.2-1

MAXIMUM DEFLECTIONS ALLOWED FOR REACTOR INTERNAL SUPPORT STRUCTURES

Component	Allowable Deflections (in)	No-Loss-Of Function Deflections (in)
Upper Barrel		
radial inward	4.1	8.2
radial outward	1.0	1.0
Upper Package	0.10	0.15
Rod Cluster Guide Tubes	1.00	1.75

TABLE 4.1-1

SYSTEM DESIGN AND OPERATING PARAMETERS

Plant design life, years	40
Number of heat transfer loops	4
Design pressure, psig	2485
Nominal operating pressure, psig	2235
Total system volume including pressurizer and surge line (ambient conditions), ft^3 (estimated)	12,500
System liquid volume, including pressurizer and surge line (ambient conditions), ft^3	11,892
System liquid volume, including pressurizer max. guaranteed power, ft^3 (estimated)	11,780
Total Reactor heat output (100% power) Btu/hr	$11,089 \times 10^6$ (Unit 1) (3250 MWt) $11,641 \times 10^6$ (Unit 2) (3411 MWt)

	Unit 1 Bounding Conditions for Rerating Lower/Upper	Unit 2
Reactor vessel coolant temperature at full power:		
Inlet, nominal, ^oF	514.9/545.2	541.3
Outlet, nominal, ^oF	579.1/607.5	606.4
Coolant temperature rise in vessel at full power, avg., ^oF	64.2/62.3	64.8
Total coolant flow rate, lb/hr x 10^6	139.0/133.9	134.6
Steam pressure at full power, psia	618/820	820
Steam Temp. @ full power, ^oF	489.4/521.1	521.1
Total Reactor Coolant Volume at ambient conditions, ft^3	12,438	12,470

TABLE 4.1-2

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	Pressure, psig	
	Unit 1	Unit 2
Design Pressure	2485	2485
Operating Pressure	2235	2235
Safety Valves	2485	2485
Power Relief Valves*	2335	2335
Pressurizer Spray Valves (Begin to Open)	2260	2260
Pressurizer Spray Valves (Full Open)	2310	2310
Pressurizer Pressure High - Reactor Trip	2378	2378
High Pressure Alarm	2310	2310
Pressurizer Pressure Low - Reactor Trip	$\geq$ 1865	$\geq$ 1950
Low Pressure Alarm	2135	2135
Pressurizer Pressure Low - Safety Injection	$\geq$ 1815	$\geq$ 1900
Hydrostatic Test Pressure	3106	3106
Backup Heaters On	2185	2185
Proportional Heaters (Begin to Operate)	2250	2250
Proportional Heaters (Full Operation)	2220	2220

*During Start-up and Shut-down when the Reactor Coolant System temperature is below 331°F for Unit 1 and 331°F for Unit 2, a safeguard circuit is manually switched on which allows opening of that Unit's two Power Relief Valves at 435 psig for Unit 1 and 435 psig for Unit 2 for low temperature overpressure protection (LTOP) of the Reactor Vessel. This safeguard circuit ensures that the reactor pressure remains below the ASME Section III, Appendix G "Protection Against Non-ductile Failure" limits in the case of an LTOP event.

The pressurizer relief tank, by means of its connection to the Waste Disposal System, provides a means for removing any non-condensable gases from the Reactor Coolant System which might collect in the pressurizer vessel.

Steam is discharged through a sparger pipe under the water level. This condenses and cools the steam by mixing it with water that is near ambient temperature. The tank is equipped with an internal spray and a drain which are used to cool the tank following a discharge. The tank is protected against a discharge exceeding the design value by two rupture discs which discharge into the reactor containment. The tank is carbon steel with a corrosion-resistant coating on the wetted surfaces. A flanged nozzle is provided on the tank for the pressurizer discharge line connection. This nozzle and the discharge piping and sparger within the vessel are austenitic stainless steel.

The tank design is based on the requirement to condense and cool a discharge of pressurizer steam equal to 110 percent of the volume above the zero-power pressurizer water level set-point. The tank is not designed to accept a continuous discharge from the pressurizer. The volume of water in the tank is capable of absorbing the heat from the assumed discharge, with an initial temperature of 120°F and increasing to a final temperature of 200°F. If the temperature in the tank rises above 120°F during plant operation, the tank is cooled by spraying in cool water and draining out the warm mixture to the Waste Disposal System.

The spray rate is designed to cool the tank from 200°F to 120°F in approximately one hour following the design discharge of pressurizer steam. The volume of nitrogen gas in the tank is selected to limit the maximum pressure following a design discharge to 50 psig.

The rupture discs on the relief tank have a relief capacity equal to the combined capacity of the pressurizer safety valves. The tank design pressure is twice the calculated pressure resulting from the maximum safety valve discharge described above. The tank and rupture discs holders are also designed for full vacuum to prevent tank collapse if the contents cool following a discharge without nitrogen being added.

Principal design parameters of the pressurizer relief tank are given in Table 4.1-4.

Discharge Piping

The discharge piping (from the safety and power-operated relief valves to the pressurizer relief tank) is sized to prevent back-pressure at the safety valves from exceeding 20 percent of the set-point pressure at full flow. The pressurizer safety and power relief valves discharge lines are stainless steel.

4.2.2.4 Steam Generators

The steam generators are vertical shell and U-tube heat exchangers with integral moisture separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head of the steam generator. The head is divided into inlet and outlet chambers by a vertical partition plate extending from the head to the tube sheet. Manways are provided for access to both sides of the divided head. Feedwater enters the steam generators and is distributed thru a feedwater ring located just below the moisture separators. Thermal sleeves are provided in the feedwater piping elbows at the steam generator inlet. Feedwater flow is out of the top of the feedwater ring thru "J" tubes, down between the steam generator shell and tube bundle wrapper and into the tube bundle just above the tube sheet. The "J" tubes prevent rapid drainage of the feedwater ring due to a drop in steam generator water

Indication of valve position for the pressurizer safety and power-operated relief valves is provided by a four channel acoustic flow monitor. There are four accelerometers, one strapped to the discharge of each of the three pressurizer safety valves and one on the common discharge of the three power relief valves. Flow through any of these valves produces an acoustic energy input to the respective accelerometer and this is amplified on the assigned channel of the monitor which is located in the control room. Indication on four vertical rows of light emitting diodes represents a bar graph display of relative flow through the monitored valves.

Pressurizer Safety Valves

The pressurizer safety valves are totally enclosed pop-type valves. The valves are spring-loaded, self-activated and with back-pressure compensation designed to prevent system pressure from exceeding the design pressure by more than 110 percent, in accordance with the ASME Boiler and Pressure Vessel Code, Section III. The set pressure of the valves is 2485 psig.

The 6" pipes connecting the pressurizer nozzles to their respective safety valves are shaped in the form of a loop seal. Piping is connected to the bottom of each loop seal to drain any condensate that accumulates in the loop seal. An acoustic flow monitor and a temperature indicator on each valve discharge alerts the operator to the passage of steam due to leakage or valve lifting.

Power Relief Valves

The pressurizer is equipped with 3 power-operated relief valves which limit system pressure for a large power mismatch and thus lessen the likelihood of an actuation of the fixed high-pressure reactor trip. The relief valves operate automatically or by remote manual control. The operation of these valves also limits the undesirable operation of the spring-loaded safety valves. Remotely operated stop valves are provided to isolate the power-operated relief valves. An acoustic flow monitor and a temperature indicator on the common discharge of the relief valves alerts the operator to the passage of steam due to leakage or valve opening.

During startup and shutdown transient conditions, when the reactor coolant system temperature is below 331°F for Unit 1 and 331°F for Unit 2, a safeguard circuit is manually energized in the control room to allow automatic opening of that unit's two power relief valves at 435 psig for Unit 1, and 435 psig for Unit 2, for low temperature over pressure (LTOP) protection of the reactor vessel. This safeguard circuit ensures that the reactor pressure remains below the ASME Section III, Appendix G "Protection Against Nonductile Failure" limits in the case of an LTOP event.

Design parameters for the pressurizer spray control, safety, and power relief valves are given in Table 4.1-8.

4.2.2.9 Reactor Coolant System Supports

1. Steam Generator Support

    Each steam generator is supported by a structural system consisting of four vertical support columns and upper and lower lateral restraints approximately 46½ feet apart. The vertical columns have a ball joint connection at each end to accommodate both the radial growth of the steam generator itself and the radial movement of the vessel from the reactor center.

The lower lateral support consists of an inner frame, keyed and shimmed to the four steam generator support feet to accommodate radial growth of these feet. The inner frame is surrounded by an outer frame which is embedded in both the primary shield and crane wall concrete. The connection between the inner and outer frame consists of a series of shimmed points which act as both guides and limit stops to allow for expansion from the center of the reactor. The lower lateral support restrains both torsional and translational movements.

The upper lateral support consists of a ring band which is shimmed to the steam generator at twelve locations around the circumference. Attached to this band are lugs 180° apart which are shimmed and guided to a structural framing system which is embedded in the crane wall and steam generator enclosure wall concrete. Hydraulic snubbers are also connected 180° apart on the band and tied to other embedded frames in a direction coincident with the direction of movement away from the reactor center. The upper lateral support restrains rapid translational movements in all horizontal directions.

2. Reactor Vessel Supports

The reactor vessel is supported by four of its eight nozzles by four individual weldments embedded in the primary shield concrete. Each nozzle pad bears on a shoe, that is supported by a heavy U-shaped weldment which wraps around the shoe. The U-shaped weldment is water-cooled at the junction of the outer flange and the web by two continuous welded angles on either side of the web. The U-shaped weldment bears vertically on two shims and is restrained horizontally by a series of shims and bearing plates. These bearing plates and shims are connected to an outer weldment which completely surrounds the U-shaped weldment and is embedded in the concrete.

The reactor support system allows the reactor to expand radially from its vertical centerline but resists rotational motion in all orthogonal planes. The nozzle horizontal centerlines translate in the vertical direction relative to the shoes.

3. Pressurizer Support

The pressurizer is supported on a ring girder which is in turn supported on a concrete slab. Horizontally, the vessel is restrained at two elevations approximately 27 feet apart.

The lower restraint consists of anchor bolts in slightly oversize holes in the ring girder. The upper restraint consists of four individual weldments embedded in concrete that allow the pressurizer to expand radially, but resist torsional and translational horizontal movements.

4. Reactor Coolant Pump Support

Each reactor coolant pump is supported vertically by three ball joint ended columns. This structural column system resists both overturning and vertical movement while allowing for expansion from the center of reactor. Excessive torsional and horizontal translational movements are resisted by a combination of lateral thrust columns anchored into the crane wall concrete.

4.2.3 PRESSURE-RELIEVING DEVICES

The Reactor Coolant System is protected against overpressure by control and protective circuits such as the high pressure trip and by relief and safety valves connected to the top head of the pressurizer. The relief and safety valves discharge into the pressurizer relief tank which condenses and collects the valve effluent. The schematic arrangement of the relief devices is shown in Figure 4.2-1A, and the valve design parameters are given in Table 4.1-8. The valves are further discussed in Sub-Section 4.2.2.7.

TABLE 4.2-3

STEAM GENERATOR WATER (STEAM-SIDE) CHEMISTRY SPECIFICATION

FOR 100% FULL POWER

Cation Conductivity[a]	≤ 0.8 umhos/cm
pH @ 25°C	8.5 - 9.4 without boric acid
	≥ 7.5 with boric acid
Chloride	≤ 20 ppb
Sodium	≤ 20 ppb
Sulfate	≤ 20 ppb

[a] If boric acid is present in the system, the cation conductivity specification will be ≤ [0.8 + 0.03 x (boron conc. in ppm)].

atmosphere, (2) the reactor coolant system, or (3) closed systems inside containment that are assumed vulnerable to accident forces. One barrier may be the containment isolation valve itself, which is between the containment atmosphere and the test connection. For closed systems inside containment, which are Seismic Class I design with a low probability of failure, only one barrier is required. Test connections are provided with locked closed valves and/or pipe caps and are administratively controlled.

10. All normally closed locked valves and caps are administratively controlled to ensure containment integrity.

NOTE: A seal may be used in lieu of a lock to satisfy the locking requirements discussed in the section above.

Containment Isolation Testing and Reliability

The containment isolation system is designed to provide such functional reliability and testing facilities as are necessary to avoid undue risk to the health and safety of the public. The air operated isolation valves close on loss of control power or air. The instrumentation and control circuits are redundant in the sense that a single failure cannot prevent containment isolation. Provision is made for periodic testing of the leak tightness and functioning of the isolation valves.

Test connections are locked closed and/or capped and are administratively controlled to ensure containment integrity. Therefore, no further testing of test connection leak tightness is required. This is consistent with the clarifications of Appendix J requirements discussed with the NRC during the CILRT inspections conducted February 9-15, 1989 (Inspection Report Nos. 50-315/89007 (DRS) and 50-316/89007 (DRS)).

Containment Isolation System Protection

Adequate protection for containment isolation, including piping, valves, and

vessels, is provided against dynamic effects and missiles which might resultfrom plant equipment failures including a loss-of-coolant accident. Isolation valves inside the containment are located between the crane wall or some other missile shield and the outside containment wall. Isolation valves, piping or vessels which provide one of the isolation barriers outside the containment are similarly protected.

Containment Isolation System Operation

No manual operation is required for immediate isolation of the containment. Automatic trip valves are provided in those lines which must be isolated immediately following an accident. Lines which must remain in service subsequent to certain accidents for safety reasons are provided with at least one remote manual valve, except instrument sensing lines that are provided with one manual valve.

Automatic trip valves may be operated by a manual switch. The position of each automatic trip valve is displayed in the main control room.

The instrumentation and controls for the system are described in more detail in Chapter 7.

Containment Isolation System Piping Classes

The functional classes of piping are used to further define the design bases.

Class A

Class A piping is open to the outside atmosphere, and is connected to the reactor coolant system, or is open to the containment atmosphere. Alternatively, Class A piping is Seismic Class III in design and is assumed to be vulnerable to accident forces.

For Class A piping the following is provided, as a minimum, for isolation subsequent to an incident:

Check valves may be employed as one of the two barriers for incoming lines. Test connections and pressurizing means are provided to test each isolation valve or barrier for leak tightness. Either water or a gas is used as the pressurizing medium depending on the requirements of each case. Where it is necessary to make a quantative leakage test, provision is made to:

a) measure the inflow of the pressurizing medium, or
b) collect and measure the leakage, or
c) calculate the leakage from the rate of pressure drop.

The test connections are isolated when not in use by locked closed manual valves and/or caps and administratively controlled to ensure containment integrity.

All isolation valves are missile protected. Isolation valves, actuators, and control devices required inside the containment are located between the missile barrier and the containment wall. Isolation valves, actuators and control devices outside the containment are located outside the path of potential missiles or provided with missile protection.

There are two levels of automatic containment isolation identified as Phase A and Phase B. Phase A isolation closes all lines penetrating the containment except essential lines such as Safety Injection and Containment Spray which are not isolated, and component cooling water to the reactor pumps and service water to the ventilation units which isolates on Phase B. (For Phase A and B initiating signals see Chapter 7 Instrumentation and Control.) All automatic isolation valves are able to be closed from the main control room. Position indicators are provided for each valve near its manual control switch in the main control room.

Specific administrative procedures govern the positioning of all isolation valves except check valves as well as any flanged closures during normal operation, shutdown and incident conditions. Check valves in incoming lines open only when the fluid pressure in the line coming from the outside is higher than the pressure on the containment side. Gravity or a spring holds the valve closed in the balanced pressure condition.

5.4.3 DESIGN EVALUATION

The containment isolation system provides two barriers to prevent leakage of radioactivity at each containment opening. Either barrier is sufficient to keep the leakage within limits.

5.4.4 TEST AND INSPECTION

All valve leak testing for Inservice Inspection (ISI) and Integrated Leak Rate Test (ILRT) program and surveillance requirements are performed in accordance with Appendix J to 10 CFR 50 for Type A, B and C type testing. Also certain valves will be tested for operability in accordance with the applicable edition of the ASME Boiler and Pressure Vessel Code, Section XI.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	Status of Isolation Valves S	Status of Isolation Valves I	Isolation Valves Inside	Isolation Valves Outside	Isolation Actuation Signal	Figure Number	Notes
Gas Analyzer From Pressurizer Relief Tank	A	1/2" (1)	Out	Int.	Closed	Closed	Auto Trip Trip	2 Auto	A	4.2-1A	
Primary Water Supply to Pressurizer Relief Tank	A	3" (1)	In	Int.	Closed	Closed	Check	Auto Trip	A	4.2-1A	
Nitrogen Supply to Pressurizer Relief Tank	A	3/4" (1)	In	Int.	Closed	Closed	Check	Auto Trip	A	4.2-1A	
Reactor Coolant Pumps Seal Water Supplies	D	2" (4)	In	Open	Open	Open	Check	-	NA	4.2-1A	
Reactor Coolant Pumps Seal Water & Excess Letdown Heat Exchanger Discharges	B	4" (1)	Out	Open	Open	Closed	Auto Trip	Auto Trip	A	9.2-1	
Reactor Coolant Pump Motor and Thermal Barrier Cooling Water Supply	B	8" (1)	In	Open	Open	Closed	-	2 Auto Trip	B	9.5-1	
Reactor Coolant Pump Motor Cooling Water Discharge	B	8" (1)	Out	Open	Open	Closed	-	2 Auto Trip	B	9.5-1	

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Letdown Line (CVCS)	B	2" (1)	Out	Open	Closed	Closed	Auto Trip	Auto Trip	A	9.2-1	
Charging Line (CVCS)	D	3" (1)	In	Open	Closed	Open	Check	-	NA	9.2-1	
Excess Letdown Heat Exchanger Component Cooling Water Inlet	C	4" (1)	In	Open	Closed	Closed	-	Auto Trip	A	9.5-1	
Excess Letdown Heat Exchanger Component Cooling Water Outlet	C	4" (1)	Out	Open	Closed	Closed	-	Auto Trip	A	9.5-1	
Reactor Coolant Drain Tank Pump Suction	A	4" (1)	Out	Int.	Int.	Closed	-	2 Auto Trip	A	11.1-1	
Containment Sump Pump Discharge to Waste Disposal	C	3" (1)	Out	Int.	Int.	Closed	-	2 Auto Trip	A	11.1-2	
Upper Containment Spray Inlet	D	8" (2)	In	Closed	Closed	Open	Check	-	NA	6.3-1	1
Lower Containment Spray Inlet	D	6" (2)	In	Closed	Closed	Open	Check	-	NA	6.3-1	1
RHR to Containment Spray	D	8" (2)	In	Closed	Closed	If Needed	Check	-	NA	6.3-1	1

1) Check valves held closed by gravity or spring in balanced pressure condition.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Residual Heat Removal Inlet to Pumps (Normal Cooldown)	B	14" (1)	Out	Closed	Open	Closed	Remote Manual	-	None	6.2-A 9.3-1	2
Residual Heat Removal To Reactor Coolant Hot Legs-Low Head S.I.	D	8" (2)	In	Open	Closed	Open	-	Remote Manual	None	6.2-A 9.3-1	
Residual Heat Removal Suction From Sump	D	18" (2)	Out	Closed	Closed	Open	-	Remote Manual	None	6.2-A 9.3-1	3
Safety Injection	D	4" (2)	In	Open	Closed	Open	-	Remote Manual	None	6.2-1	
Safety Injection Test Line and Accumulator Test Line	A	3/4" (1)	In or Out	Int.	Closed	Closed	-	2 Manual (L.C.)	NA	-	
Boron Injection Inlet	D	3" (1)	In	Closed	Closed	Open	-	Remote Manual	None	6.2-1	4
Residual Heat Removal to Reactor Coolant Cold Legs (Normal Cooldown)	B	12" (1)	In	Closed	Open	If Needed	Remote Manual	-	None	6.2-A 9.3-1	

2) Valve administratively locked closed.
3) Open during recirculation mode.
4) Open automatically on Safety Injection Signal.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Nitrogen to Accumulators	A	1" (1)	In	Int.	Int.	Closed	Check	Auto Trip	A	6.2-A 9.3-1	
Sample Line From Pressurizer Steam Space	A	1/2" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	9.6-1	
Sample Line from Pressurizer Liquid Space	A	1/2" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	9.6-1	
Sample Line from Hot Legs	A	1/2" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	9.6-1	
Sample Lines from Accumulators	A	1/2" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	9.6-1	
Sample Lines from Steam Generator Steam Outlets	C	1/2" (4)	Out	Open	Closed	Closed	-	Auto Trip	A	9.6-1	
Steam Generator Main Steam Outlets	C	30" (4)	Out	Open	Closed	Closed	-	-	B	10.2-1	5
Steam Generator Blowdown Lines	C	2" (4)	Out	Int.	Closed	Closed	-	Auto Trip	A	10.2-1	

5) Steam Generator Stop Valves located outside containment also close on steamline isolation signal as described in Chapter 7.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	Status of Isolation Valves S	Status of Isolation Valves I	Isolation Valves Inside	Isolation Valves Outside	Isolation Actuation Signal	Figure Number	Notes
Steam Generator Feedwater Supply	C	14" (4)	In	Open	Closed	Closed	-	Check	NA	10.5-1	
Steam Generator Auxiliary Feedwater Supply	C	6" (4)	In	Open	Int.	If Needed	-	Check	NA	10.5-1	6
Steam Generator Chemical Feed Supply	C	1/2" (4)	In	Closed	Int.	Closed	-	Check	NA	10.5-1	6
Non Essential Service Water to Containment Ventilation Units	A	6" (4) 3" (4)	In	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Non Essential Service Water from Containment Ventilation Units	A	6" (4) 3" (4)	Out	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Purge Air Inlet (Containment)	A	30" (1) 24" (1)	In	If Needed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	
Purge Air Outlet (Containment)	A	30" (1) 24" (1)	Out	If Needed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	

6) No independent containment penetrations. These lines join the Feedwater Lines between the penetrations and the isolation valves.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines		Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Fuel Transfer Tube	A	20"	(1)	In or Out	Closed	Open	Closed	Blind Flange	-	NA	-	7
Service Air	A	2"	(1)	In	Closed	Open	Closed	Check	Auto Trip	A	9.8-3	
Instrument Air	A	1"	(2)	In	Open	Open	Closed	-	2 Auto Trip	A	9.8-3	
Reactor Coolant Pump Thermal Barrier Cooling Water Discharge	B	4"	(1)	Out	Open	Open	Closed	-	2 Auto Trip	B	9.5-1	
Gas Analyzer From Reactor Coolant Drain Tank	A	1/2"	(1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	11.1-1	
Ice Loading Line	E	5"	(1)	In	Closed	If Needed	Closed	Blind Flange	Blind Flange	NA	5.3-2A	
Containment Pressure Relief Line	A	12"	(1)	Out	If Needed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	
Containment Test Pressurization	E	5"	(1)	In	Closed	If Needed	Closed	Blind Flange	Blind Flange	NA	9.8-3	8

7) See Sub-Chapter 5.2 for description of double gasketed seal on the Fuel Transfer Tube.
8) Same physical line as ice loading line.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Ice Loading Return	E	5" (1)	Out	Closed	Int.	Closed	Blind Flange	Blind Flange	NA	5.3-2A	
Glycol to Ice Condenser Fan Coolers	E	3" (1)	In	Open	Open	Closed	Auto Trip	Auto Trip	A	5.3-2A	
Glycol from Ice Condenser Fan Coolers	E	3" (1)	Out	Open	Open	Closed	Auto Trip	Auto Trip	A	5.3-2A	
Bypass Glycol line to Ice Condenser Fan Coolers	E	3/8" (1)	In	Open	Open	Closed	Check	-	NA	5.3-2A	
Bypass Glycol line from Ice Condenser Fan Coolers	E	3/8" (1)	Out	Open	Open	Closed	Check	-	NA	5.3-2A	
Purge Air Inlet (Instrumentation Room)	A	14" (1)	In	Closed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	8a
Purge Air Outlet (Instrumentation Room)	A	14" (1)	Out	Closed	If Needed	Closed	Auto Trip	Auto Trip	A or CVI	5.5-2	8a
Reactor Coolant Drain Tank & Press. Relief Tank Vents	B	1" (1)	Out	Int.	Closed	Closed	-	2 Auto Trip	A	11.1-1	

8a) For status "N": "Closed" for Unit 2; Unit 1 is "If needed" (for limited purging).

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Refueling Water Supply	A	2 1/2" (1)	In	Closed	Int.	Closed	-	2 Manual (L.C.)	NA	9.4-1	
Demineralized Water Supply	A	2" (1)	In	Closed	Open	Closed	-	2 Auto Trip	A	-	
Non Essential Service Water to Reactor Coolant Pump Motor Air Coolers	A	3" (4)	In	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Non Essential Service Water from Reactor Coolant Pump Motor Air Coolers	A	3" (4)	Out	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Reactor Support Cooling Inlet	C	2 1/2" (1)	In	Open	If Needed	Closed	Check	Auto Trip	A	9.5-1	
Reactor Support Cooling Outlet	C	2 1/2" (1)	Out	Open	If Needed	Closed	-	2 Auto Trip	A	9.5-1	
Refueling Cavity Drain To Purification System	A	3" (1)	Out	Closed	If Needed	Closed	-	2 Manual (L.C.)	NA	11.1-1	
Nitrogen Supply to Reactor Coolant Drain Tank	A	1" (1)	In	Open	Open	Closed	-	Auto Trip & Check	A	11.1-1	9

9) No independent containment penetration. Joins RCDT vent line between penetration and isolation valves.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Steam Generator Blowdown Samples	C	1/2" (4)	Out	Int.	Closed	Closed	-	Auto Trip	A	9.6-1	
Containment Weld Channel Pressurization Air Supply	D	1/2" (2)	In	Closed	If Needed	Open	-	Check	NA	5.6-1	10
Dead Weight Test Connection	E	1/2" (1)	-	Int.	Closed	Closed	-	Manual	NA	4.2-1A	
Relief Vent Header	B	4" (1)	In	Int.	Int.	Int.	Check	-	NA	4.2-1A	
Ice Condenser and Containment Ventilation Unit Drain to Drain Header	A	3" (1) 1" (1)	Out	Open	Open	Closed	-	2 Auto Trip (Each Line)	A	11.1-1	
Component Cooling Water to Main Steam Penetrations	D	1" (4)	In	Open	Open	Open	Check	Manual	NA	-	
Component Cooling Water from Main Steam Penetrations	D	1 1/2" (2)	Out	Open	Open	Open	-	Remote Manual	None	-	
Component Cooling Water to Pressure Equalizing Fans	D	1 1/2" (2)	In	Closed	Closed	Open	-	Remote Manual	None	-	

10) May be used for Leak Test of Channels.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Component Cooling Water from Pressure Equalizing Fans	D	1 1/2" (2)	Out	Closed	Closed	Open	-	Remote Manual	None	-	
Containment Air Particulate and Radio Gas Detector Sample Line	B	1" (2)	Out	Open	Open	Int.	-	2 Auto Trip	B	-	11
Containment Air Particulate and Raido Gas Detector Sample Return	B	1" (1)	In	Open	Open	Int.	Check	Auto Trip	B	-	11
Lower Containment Radiation Sampling System	A	1/2" (2)	Out	If Needed	Closed	Closed	-	2 Manual	NA	-	
Upper Containment Radiation Sampling System	A	1/2" (2)	Out	If Needed	Closed	Closed	-	2 Manual	NA	-	
Instrument Room Radiation Sampling System	A	1/2" (2)	Out	If Needed	Closed	Closed	-	2 Manual	NA	-	

11) May be put in service manually after incident

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	S	I	Isolation Valves Inside	Outside	Isolation Actuation Signal	Figure Number	Notes
Non Essential Service Water to Instrument Ventilation Units	A	2 1/2" (2)	In	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Non Essential Service Water from Instrument Room Ventilation Units	A	2 1/2" (2)	Out	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Sample Lines to Hydrogen Monitoring System	D	1/2" (9)	Out	Closed	Closed	Int.	-	2 Auto Trip	A	14.3.6-12A	11
Sample Line Return From Hydrogen Monitoring System	D	1/2" (1)	In	Closed	Closed	Int.	-	Auto Trip	A	14.3.6-12A	11
Containment Pressure Transmitters	E	1/2" (6)	-	Open	Open	Open	-	Manual	NA	-	12
Containment Sump Sample to Post-Accident Sampling System	D	1/2" (1)	Out	Closed	Closed	Int.	-	Auto Trip	A	9.6-2	11
Post Accident Sampling System Return	D	1/2" (1)	In	Closed	Closed	Int.	Check	Auto Trip	A	9.6-2	11

11) May be put in service manually after incident
12) See Fig. 7.5-1 for a functional diagram of these instruments.

TABLE 5.4-1
PIPING PENETRATIONS

Service	Class	Line Size and Number of Lines	Flow Direction	Status of Isolation Valves N	Status of Isolation Valves S	Status of Isolation Valves I	Isolation Valves Inside	Isolation Valves Outside	Isolation Actuation Signal	Figure Number	Notes
Post Accident Sampling System Supply (Gas)	D	1/2" (1)	Out	Closed	Closed	Int.	-	2 Auto Trip	A	9.6-2	13
Reactor Vessel Level Instrumentation System	E	3/16" (6)	-	Open	Open	Open	-	Membrane Barrier	NA	-	
Incore Flux Detection System	NA	8" (1)	-	Closed	If Needed	Closed	Blind Flange	Blind Flange	NA	-	14
Spare Penetrations	NA	18" (4) 6" (4)	-	Closed	Closed	Closed	Weld Cap	Weld Cap	NA	-	
Service Penetration	E	18" (1)	-	Closed	If Needed	Closed	Hinged Closures	Hinged Closures	NA	-	

13) Connected to Containment Air Particulate and Radio Gas Detector Sample Line
14) Used for replacement of incore flux instrumentation thimbles.

N: Normal
S: shutdown
I: Incident

Int: Intermittent
L.C.: Locked Closed
NA: Not Applicable

Isolation Actuation Signals:
A: Phase A Isolation
B: Phase B Isolation
CVI: Containment Ventilation Isolation (initiated by Safety Injection Signal or High Containment Radiation)