REGULATO. INFORMATION DISTRIBUTION ~YSTEM (RIDS)

ACCESSION NBR: 8707300258 DOC. DATE: 87/07/22 NOTARIZED: NO DOCKET #
FACIL: 50-315 Donald C. Cook Nuclear Power Plant, Unit 1, Indiana & 05000315
50-316 Donald C. Cook Nuclear Power Plant, Unit 2, Indiana & 05000316

AUTH. NAME	AUTHOR AFFILIATION
ALEXICH, M. P.	Indiana & Michigan Electric Co.
RECIP. NAME	RECIPIENT AFFILIATION
MURLEY, T. E.	NRC - No Detailed Affiliation Given

SUBJECT: Forwards 1987 version of FSAR for DC Cook Nuclear Plant, Units 1 & 2. Rev 8/10/87 w

DISTRIBUTION CODE: A053D COPIES RECEIVED: LTR 1 ENCL 10 SIZE: 1+1,200
TITLE: OR Submittal: Updated FSAR (50.71) and Amendments

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD3-3 LA	1	0	PD3-3 PD	1	0
	WIGGINGTON, D	1	1			
INTERNAL:	AEOD/DOA/IRB	1	1	~~NRR/DEST/ADE~~	1	0
	NRR/DEST/ADS	1	0	REG FILE 01	1	1
	RES/DE/EIB	1	1	RGN3	1	1
EXTERNAL:	DMB/DSS	1	1	FRANKLIN	1	1
	LPDR	1	1	NRC PDR	1	1
	NSIC	1	1			

TOTAL NUMBER OF COPIES REQUIRED: LTTR 14 ENCL 10

INDIANA & MICHIGAN ELECTRIC COMPANY

P.O. BOX 16631
COLUMBUS, OHIO 43216

July 22, 1987

AEP:NRC:0509I
10 CFR 59.71(e)

Donald C. Cook Nuclear Plant
Docket Nos. 50-315 and 50-316
License Nos. DPR-58 and DPR-74
1987 FSAR UPDATE

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

Attn: T. E. Murley

Dear Dr. Murley:

We are transmitting to you under separate cover ten (10) copies of the changed pages for the 1987 version of the Final Safety Analysis Report (FSAR) for the Donald C. Cook Nuclear Plant, Unit Nos. 1 and 2. These pages are being transmitted to you according to the provisions of 10 CFR 50.71(e). A list of replacement pages is included with each copy.

Changed pages have been dated "July, 1987" in the lower right corner in order to maintain a reference point for changed pages in addition to vertically barring the specific change.

We hereby certify that the information contained in this update to the FSAR, to the best of our knowledge, accurately presents changes made since the previous submittal.

Very truly yours,

M. P. Alexich
Vice President

MPA/naw

cc: John E. Dolan
W. G. Smith, Jr. - Bridgman
R. C. Callen
G. Bruchmann
G. Charnoff
NRC Resident Inspector - Bridgman

A053
1/10

Remove Pages	Insert Pages	Add New Page
TABLE OF CONTENTS		
3-iii to 3-vi (Unit 2)	3-iii to 3-vi (Unit 2)	
3-xi (Unit 2)	3-xi (Unit 2)	
14-i to 14-xxxvi (Unit 2)	14-i to 14-xxvi (Unit 2)	
CHAPTER 1		
1.4-1 to 1.4-20	1.4-1 to 1.4-20	
Table 1.4-1 (pg 1)	Table 1.4-1 (pg 1)	
1.6-18 to 1.6-19	1.6-18 to 1.6-19	
Table 1.6-1 (last pg)	Table 1.6-1 (last pg)	
1.7-1 to 1.7-92	1.7-1 to 1.7-88	
1.7.A-93 to 1.7.A-96	1.7.A-89 to 1.7.A-97	
1.7.B-97 to 1.7.B-115	1.7.B-98 to 1.7.B-116	
CHAPTER 2		
2.2-3 to 2.2-14	2.2-3 to 2.2-14	
2.7-5 to 2.7-6	2.7-5 to 2.7-6	
Table 2.7-3	Table 2.7-3	
Table 2.7-4	Table 2.7-4	
Figure 2.7-2	Figure 2.7-2	
CHAPTER 3		
UNIT 1		
3.2-45 to 3.2-50	3.2-45 to 3.2-50	
3.3-11 to 3.3-12	3.3-11 to 3.3-12	
3.3-25 to 3.3-26	3.3-25 to 3.3-26	
3.5-27 to 3.5-30	3.5-27 to 3.5-30	
		Table 3.6.2-4
3.6-13 to 3.6-16	3.6-13 to 3.6-16	
Table 3.6.1-3 (1986)	Table 3.6.1-3 (1985)	
Table 3.6.2-2	Table 3.6.2-2	

Remove Pages	Insert Pages	Add New Page
UNIT 2		
Table 3.1-1	Table 3.1-1	
Table 3.3-1	Table 3.3-1	
3.5-1 to 3.5-25	3.5-1 to 3.5-20	
Table 3.5.1-1	Table 3.5.1-1	
Table 3.5.1-2 (2 pp)	Table 3.5.1-2 (2 pp)	
Table 3.5.1-3	Table 3.5.1-3	
Fig. 3.5.1-1	Fig. 3.5.1-1	
Fig. 3.5.1-3 to Fig. 3.5.1-4	Fig. 3.5.1-3 to Fig. 3.5.1-4	
3.5-35 to 3.5-44	3.5-21 to 3.5-29	
Table 3.5.2-1	Table 3.5.2-1	
Table 3.5.2-2	Table 3.5.2-2	
Table 3.5.2-3	Table 3.5.2-3	
Fig. 3.5.2-1 to Fig. 3.5.2-8	Fig. 3.5.2-1 to Fig. 3.5.2-7	
3.5-56 to 3.5-58	3.5-30 to 3.5-32	
Table 3.5.3-1	Table 3.5.3-1	
CHAPTER 4		
Table 4.1-1	Table 4.1-1	
Table 4.1-5	Table 4.1-5	
4.2-33 to 4.2-35	4.2-33 to 4.2-36	
CHAPTER 5		
5.1-1 to 5.1-2	5.1-1 to 5.1-2	
5.2-1 to 5.2-2	5.2-1 to 5.2-2	
5.2-27 to 5.2-28	5.2-27 to 5.2-28	
5.2-31 to 5.2-115	5.2-31 to 5.2-115	
5.4-5 to 5.4-6	5.4-5 to 5.4-6	
Table 5.4-1 (pp 11 & 12 of 12)	Table 5.4-1 (pp 11 & 12 of 12)	

Remove Pages	Insert Pages	Add New Page
CHAPTER 6		
6.1-1 to 6.1-4	6.1-1 to 6.1-4	
Table 6.1-1	Table 6.1-1	
CHAPTER 7		
7.2-25 to 7.2-26	7.2-25 to 7.2-26	
7.2-29 to 7.2-54	7.2-29 to 7.2-54	
Table 7.2-1 (5 pgs)	Table 7.2-1 (5 pgs)	
Table 7.2-2 (sheet 4 of 4) to Table 7.2-3	Table 7.2-2 (sheet 4 of 4) to Table 7.2-3	
7.3-1 to 7.3-2	7.3-1 to 7.3-2	
7.4-1 to 7.4-2	7.4-1 to 7.4-2	
7.5-7 to 7.5-10	7.5-7 to 7.5-10	
7.5-19 to 7.5-20	7.5-19 to 7.5-20	
Table 7.5-3 (sheet 2 of 2) to Table 7.5-4	Table 7.5-3 (sheet 2 of 2) to Table 7.5-4	
CHAPTER 8		
8.1-5 to 8.1-8	8.1-5 to 8.1-8	
8.3-3 to 8.3-4	8.3-3 to 8.3-4	
CHAPTER 9		
9.2-7 to 9.2-8	9.2-7 to 9.2-8	
9.3-3 to 9.3-4	9.3-3 to 9.3-4	
9.3-9 to 9.3-12	9.3-9 to 9.3-12	
9.5-1 to 9.5-2	9.5-1 to 9.5-2	
9.6-1 to 9.6-6	9.6-1 to 9.6-6	
9.7-5 to 9.7-6	9.7-5 to 9.7-6	

Remove Pages	Insert Pages	Add New Page
9.7-9 to 9.7-21	9.7-9 to 9.7-21	
9.8-1 to 9.8-28	9.8-1 to 9.8-30	
Table 9.8-4 (pp. 2 & 3)	Table 9.8-4 (pp 2 & 3)	
Table 9.8-5	Table 9.8-5	
Fig. 9.8-3	Fig. 9.8-3	
Fig. 9.8-4	Fig 9.8-4	
Fig. 9.8-5	Fig. 9.8-5	
9.9-1 to 9.9-2	9.9-1 to 9.9-2	
Fig. 9.8-3 to Fig. 9.8-5	Fig. 9.8-3 to Fig. 9.8-5	
9.10-1 to 9.10-4	9.10-1 to 9.10-4	

CHAPTER 10

10.1-1 to 10.1-2	10.1-1 to 10.1-2	
10.3-5 to 10.3-6	10.3-5 to 10.3-6	
10.4-1 to 10.4-2	10.4-1 to 10.4-2	
10.5-3 to 10.5-4	10.5-3 to 10.5-4	

CHAPTER 11

11.1-9 to 11.1-10	11.1-9 to 11.1-10	
Table 11.1-1 to Table 11.1-2	Table 11.1-1 to Table 11.1-2	
Table 11.1-4 to Table 11.1-5	Table 11.1-4 to Table 11.1-5	
11.3-5 to 11.3-14	11.3-5 to 11.3-14	
Table 11.3-1 (3 pp)	Table 11.3-1 (3 pp)	
11.4-1 to 11.4-12	11.4-1 to 11.4-12	
11.5-1 to 11.5-2	11.5-1 to 11.5-2	
11.6-1 to 11.6-4	11.6-1 to 11.6-4	
Fig 11.6-3	Fig. 11.6-3	

CHAPTER 12

12.1-1 to 12.1-2	12.1-1 to 12.1-2	

Remove Pages	Insert Pages	Add New Pages
CHAPTER 14		
Unit 1		
14.0-1 to 14.0-2	14.0-1 to 14.0-2	
14.1-3 to 14.1-4	14.1-3 to 14.1-4	
14.1.13-15 to 14.1.13-16	14.1.13-15 to 14.1.13-16	
Table 14.1.13-1	Table 14.1.13-1	
14.2-58 to 14.2-62	14.2.58 to 14.2-62	
Fig. 14.2.7-3 to Fig. 14.2.7-4	Fig. 14.2.7-3 to Fig. 14.2.7-4	
Fig. 14.2.7-11 to Fig. 14.2.7-12	Fig.2.7-11 to Fig. 14.2.7-12	
14.3.2-1 to 14.3.2-6	14.3.2-1 to 14.3.2-6	
14.3.3-1 to 14.3.3-11	14.3.3-11	
14.3.5-25 to 14.3.5-37	14.3.2-25 to 14.3.5-32	
Table 14.3.5-9 (3 pp.)	Table 14.3.5-9 (6 pp.)	
Table 14.3.5-10	-----------	
Table 14.3.5-11	-----------	
Fig. 14.3.5-5 to Fig. 14.3.5-6	Fig. 14.3.5-5 to Fig. 14.3.5-6	
14.4.2-1 to 14.4.2-2	14.4.2-1 to 14.4.2-2	
14.4.2-13 to 14.4.2-14	14.4.2-13 to 14.4.2-14	
14.4.6-1 to 14.4.11-3	14.4.6-1 to 14.4.11-8	
Table 14.4.6-5	Table 14.4.6-5	
		Table 14.4.6-5a
		Table 14.4.6-5b
		Table 14.4.6-5c
		Table 14.4.11-2 to Table 14.4.11-9
14.A-12 to 14.A-17	14.A-12 to 14.A-17	
14.A-21 to 14.A-22	14.A-21 to 14.A-22	
14C-11 to 14C-12	14C-11 to 14C-12	
14C-21 to 14C-62	14C-21 to 14C-62	
		Appendix G (eleven pages)

Remove Pages	Insert Pages	Add New Pages
Unit 2		
Sections 14.0 to 14.2.8 (Volume IX) including tables and figures	14.0.1-1 to 14.2 References	
14.3.1-1 to 14.3-34	14.3.1-1 to 14.3.1-7 and References	
Table 14.3.1-1 to Table 14.3.1-19	Table 14.3.1-1 to Table 14.3.1-6	
Fig. 14.3.1-1 to Fig. 14.3.1-135	Fig. 14.3.1-1 to Fig. 14.3.1-7	
Table 14.3.4-35 to Table 14.3.4-36	Table 14.3.4-35 to Table 14.3.4-36	
14.3.5-3 to 14.3.5-6	14.3.5-3 to 14.3.5-7	
Table 14.3.5-2 (pg. 2) to Table 14.3.5-6	Table 14.3.5-2 (pg. 2) to Table 14.3.5-7	
14.4 (one page)	14.4-1 to Table 14.8.3-3	
14.A (one Page)	14.A-1	
Appendix 14C (including tables and figures)	------------	

A.5 REACTOR COOLANT TRITIUM SOURCES

GENERAL DISCUSSION

During the fissioning of uranium, tritium atoms are generated in the fuel at a rate of approximately 8×10^{-5} atoms per fission (1.05×10^{-2} curies/mwt - day). Other sources of tritium include neutron reactions with boron (in the coolant for shim control), neutron reactions with lithium (utilized in the coolant for pH control, and produced in the coolant by neutron reactions with boron), and by neutron reactions with naturally occurring deuterium in light water. The source term data is presented in Tables 14A.5-1 and 14A.5-2.

A. <u>Release of Ternary Produced Tritium</u>

The tritium formed by ternary fission in uranium fueled reactors can be retained in the fuel, accumulate in the void between the fuel and cladding, react with cladding material (zirconium tritide), or diffuse through the cladding into the coolant. Operating experience at the Shippingport reactor (zircaloy clad) indicated that less than 1% of the ternary produced tritium is released to the reactor coolant. In order to insure adequate sizing of liquid waste treatment facilities, WNES conservatively assumes that 30% of the ternary produced tritium is released to coolant. This assumption then requires that the waste treatment system be sized to process approximately 4 reactor coolant system volumes in addition to normal reactor plant liquid wastes. Anticipated ternary tritium loss to the reactor coolant is 1%.

B. <u>Tritium Produced from Boron Reactions</u>

The neutron reactions with boron resulting in the production of tritium are:

B^{10} (n, 2α) T
B^{10} (n, α), Li^7 (n, n α) T
B^{11} (n, T) Be^9
B^{10} (n, d) Be^9 (n, α) T

Of the above reactions, only the first two contribute significantly to the tritium production. The B^{11} (n, T) Be^9 reaction has a threshold of 14 Mev and a cross section of ∿ 5 mb, since the number of neutrons produced at this energy are less than 10^9 n/cm^2 -sec the tritium produced from this reaction is negligible. The B^{10} reaction may be neglected, since Be^9 has been found to be unstable.

C. Tritium Produced from Lithium Reactions

The neutron reactions with lithium resulting in the production of tirtium are:

Li^7 (n, nα) T
Li^6 (n, α) T

In the WNES designed reactors, lithium is used to maintain the reactor coolant pH at ∿ 9.5. The reactor coolant is maintained at a maximum level of 2.2 ppm lithium. A cation demineralizer is included in the Chemical and Volume Control System to remove the excess lithium produced in the B^{10} (n, α) Li^7 reactions.

The Li^6 (n, α) T reaction is controlled by limiting the Li^6 impurity in the lithium used in the reactor coolant and in lithiating the demineralizers to less than 0.001 parts of Li^6. This limitation has been in effect on WAPD designed reactors since 1962.

D. Tritium Production from Deuterium Reactions

Since the amount of naturally occurring deuterium is less than 0.00015 the tritium produced from this reaction is negligible; less than 1 curie per year.

E. Tritium Sources from the Reactor Employing Ag-In-Cd Absorber Rods

Basic Assumptions and Plant Parameters:

1.	Core thermal power	3391 MWt
2.	Plant load factor	0.8
3.	Core volume	1153 ft^3
4.	Core volume fractions	
	a. UO_2	.3052
	b. Zr + SS	.1000
	c. H_2O	.5948
5.	Initial reactor coolant boron level	
	a. Initial cycle	840 ppm
	b. Equilibrium cycle	1200 ppm
6.	Reactor coolant volume	12,560 ft^3
7.	Reactor coolant transport times	
	a. In-core	0.77 sec
	b. Out-of-core	10.87 sec
8.	Reactor coolant peak lithium level (99.9% pure Li^7)	2.2 ppm
9.	Core averaged neutron fluxes:	n/cm^2-sec
	a. E > 6 Me	2.91×10^{12}
	b. E > 5 Mev	7.90×10^{12}
	c. 3 Mev $\leq$ E $\leq$ 6 Mev	2.26×10^{13}
	d. 1 Mev $\leq$ E $\leq$ 5 Mev	5.31×10^{13}
	e. E < 0.625 ev	2.26×10^{13}

10. Neutron reaction cross-sections

a. B^{10} (n, 2α) T: σ (1 Mev $\leq$ E $\leq$ 5 Mev) = 31.6 mb (spectrum weight)

σ (E > 5 Mev) = 75 mb

b. Li^{7} (n, nα V) T: σ (3 Mev $\leq$ E $\leq$ 6 Mev) = 39.1 mb (spectrum weight)

σ (E > 6 Mev) = 400 mb

11. Fraction of ternary tritium diffusing through zirconium cladding

a. Design value 0.30

b. Expected value 0.01

F. Revised Tritium Source Term Data

Because of the importance of the ternary fission source on the operation of the plant, Westinghouse has been closely following operating plant data. A program is being conducted at the R. E. Ginna Plant to follow this in detail. The R. E. Ginna Plant has a zircaloy clad core with silver-indium-cadmium control rods. The operating levels of boron concentration during the startup of the plant are approximately 1100 to 1200 ppm of boron. In addition, burnable poison rods in the core contain boron which will contribute some tritium to the coolant, but only during the first cycle. Data during the operation of the plant has indicated very clearly that the present design sources were indeed conservative. The tritium released is essentially from the boron dissolved in the coolant and a ternary fission source which is less than ten percent. In addition to this data, other operating plants with zircaloy clad cores have also reported low tritium concentrations in the reactor coolant system after considerable periods of operation.

The revised tritium source term data developed as a result of this program is presented in Table 14A.5-2.

TABLE 14A.5-1

TRITIUM PRODUCTION IN THE REACTOR COOLANT (ci/yr)*

Tritium Source	Total Produced	Released to the Coolant: Design Value	Released to the Coolant: Expected Value
Ternary Fissions	10,420	3126	104
Burnable Poison Rods (Initial Cycle)	922	277	9
Soluble Poison Boron			
(Initial Cycle)	378	378	378
(Equilibrium Cycle)	525	525	525
Li-7 Reaction	11	11	11
Li-6 Reaction	6	6	6
Deuterium Reaction	1	1	1
Totals Initial Cycle	11,738	3799	509
Totals Equilibrium Cycle	10,963	3669	647

*This table was applicable at the time Unit 1 was licensed.

TABLE 14.A.5-2

Revised Tritium Production In The Reactor Coolant*

Tritium Source	Total Produced (curies/yr)	Expected Release to Reactor Coolant (curies/yr)
Ternary Fission	10,000	1000
Burnable Poison Rods (Initial Cycle)	1420	142
Soluble Boron		
(Initial Cycle)	206	206
(Equilibrium Cycle)	294	294
Lithium and Deuterium Reactions	105	105
Total Initial Cycle	11,730	1453
Total Equilibrium Cycle	10,400	1400

Basis:

Release Fraction from Fuel	10%
Release Fraction from Burnable Poison Rods	10%
Burnable Poison Rod B-10 Mass	6160 gpm

*This table was included in the Original FSAR in May, 1976.

TABLE 14A.8-1

CONCENTRATION OF IODINE
ISOTOPES IN THE RECIRCULATION LOOP

Isotope	Recirculation Loop Concentration (c/cc)
I-131	1.06×10^3
I-132	1.83×10^2
I-133	8.26×10^2
I-134	1.96×10^2
I-135	4.08×10^2

The radiation sources circulating in the residual heat removal loop are shown in Table 14A.8-2 and are used for whole body radiation doses in the auxiliary building.

The radioactivity in the containment also would be additional source of radiation to the auxiliary building following a loss-of-coolant accident.

TABLE 14.A.8-2

RADIATION SOURCES CIRCULATING IN RESIDUAL HEAT
REMOVAL LOOP AND ASSOCIATED EQUIPMENT - MEV/CC - SEC

DECAY TIME	GAMMA ENERGY (MEV/PHOTON) Ø.4	Ø.8	1.3	1.7	2.2	2.5	3.5
Ø HR	1.63E+Ø7	1.31E+Ø8	8.54E+Ø6	4.90E+Ø6	4.61E+Ø6	1.7ØE+Ø6	4.50E+Ø5
Ø.5 HR	1.51E+Ø7	1.23E+Ø8	7.56E+Ø6	4.16E+Ø6	4.16E+Ø6	1.61E+Ø6	3.78E+Ø5
1 HR	1.39E+Ø7	1.14E+Ø8	6.18E+Ø6	3.46E+Ø6	3.67E+Ø6	1.2ØE+Ø6	2.78E+Ø5
2 HR	1.28E+Ø7	1.Ø3E+Ø8	4.59E+Ø6	2.53E+Ø6	3.Ø1E+Ø6	8.24E+Ø5	2.ØØE=Ø5
8 HR	1.11E+Ø7	7.75E+Ø7	7.16E+Ø5	4.16E+Ø5	5.61E+Ø5	1.3ØE+Ø5	2.51E+Ø4
1 DY	1.Ø3E+Ø7	6.99E+Ø7	4.84E+Ø4	1.82E+Ø4	1.75E+Ø5	7.Ø7E+Ø3	9.96E+Ø1
1 WK	9.54E+Ø6	4.88E+Ø7	1.16E+Ø2	2.93E+Ø2	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ
1 MO	1.21E+Ø6	4.69E+Ø7	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ
6 MO	4.16E+Ø4	1.56E+Ø7	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ
1 YR	1.22E+Ø3	1.31E+Ø7	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ	Ø.ØØE+ØØ

Superseded per ltr pgs to
UFSAR dtd 7/22/87 # 8707300258
50-315

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

CHAPTER 3

TABLE OF CONTENTS (Cont'd)

CHAPTER 3

LIST OF TABLES

CHAPTER 3

LIST OF TABLES (Cont'd)

CHAPTER 3

LIST OF FIGURES (Cont'd)

TABLE OF CONTENTS

CHAPTER 14

TABLE OF CONTENTS

CHAPTER 14

TABLE OF CONTENTS (Cont'd)

CHAPTER 14

TABLE OF CONTENTS (Cont'd)

CHAPTER 14

TABLE OF CONTENTS (Cont'd)

CHAPTER 14

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

CHAPTER 14

TABLE OF CONTENTS (Cont'd)

CHAPTER 14

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

CHAPTER 14

TABLE OF CONTENTS (Cont'd)

CHAPTER 14

LIST OF TABLES

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

LIST OF TABLES (Cont'd)

CHAPTER 14

LIST OF FIGURES

LIST OF FIGURES (Cont'd)

CHAPTER 14

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

CHAPTER 14

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 14

LIST OF FIGURES (Cont'd)

LIST OF FIGURES (Cont'd)

1.4 GENERAL DESIGN CRITERIA

The general design criteria followed in the design of this plant have been developed as performance criteria which define or describe safety objectives and procedures, and they provide a guide to the type of plant design information which is included in this report. These criteria are specifically addressed in the chapters of the FSAR where they are pertinent. An index to the criteria is given in Table 1.4-1. In the chapter where a specific criterion is relevent to the design, the criterion is quoted and is followed by a brief summary of the design or procedures. The design or procedures are then more fully described in other sections of the chapter. Other criteria which apply generally to the design of the plant are given in Section 1.4.1.

In addition, the Donald C. Cook Nuclear Plant has been designed to comply with the Applicant's understanding of the intent of the AEC proposed General Design Criteria, as published for comment by the AEC in July, 1967.(1) The application of the AEC proposed General Design Criteria to the Donald C. Cook Nuclear Plant was discussed in the original FSAR, Appendix H. Table 1.4-1 contains a cross-index between the AEC design criteria and the FSAR chapters where those criteria are interpreted.

1.4.1 OVERALL PLANT REQUIREMENTS

Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication, and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary.

Quality assurance programs, test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures and inspection acceptance criteria used is required. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.

Those features of the reactor facility which are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences were designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. See Sub-Chapter 1.7 for a discussion of the quality assurance program. Recognized codes and standards were used when appropriate to the application.

Features of the facility essential to accident prevention and mitigation, are the fuel, reactor coolant system and containment barriers; the controls and emergency cooling system, whose function is to maintain the integrity of these three barriers; systems which depressurize and reduce the contamination level of the containment; power supplies and essential services to the above features; and the components employed to safely convey and store radioactive wastes and spent reactor fuel. Quality standards for material selection, design, fabrication, and inspection governing the above features conform to the applicable provisions of recognized codes and good nuclear practice.

<u>Performance Standards</u>

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or to the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated, and erected to performance standards that enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded at the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

Those features of the reactor facility which are essential to the prevention of accidents which could affect the public health and safety or to the mitigation of their consequences were designed, fabricated, and erected to performance standards that enable the facility to withstand, without loss of the capability to protect the public, the additional forces imposed by the most severe earthquakes, flooding conditions, winds, ice, or other natural phenomena characteristic to the Donald C. Cook Nuclear Plant site.

Piping, components and supporting structures of the reactor and safety related systems were designed to withstand any seismic disturbance predictable for the site. The dynamic response of the structure to ground acceleration, based on appropriate spectral characteristics of the site foundation and on the damping of the foundation and structure, was included in the design analysis.

Structures, equipment, and piping materials, in both the containment and auxiliary buildings, have been selected for their compatibility with the expected normal and accident environments. For those components located inside the containment which are required for controlling the Design Bases Accidents (DBA), the effect of the spray chemical additive (NaOH) has been considered as well as radiation levels, pressure and temperature. Material compatibility has been discussed in detail in the Indian Point Unit 2 FSAR (reference document 50-247).

Fire Protection

Criterion: A reactor facility shall be designed to ensure that the probability of events such as fires and explosions and the potential consequences of such events will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility wherever necessary to preclude such risk, particularly in areas containing critical portions of the facility such as containment, control room, and components of engineered safety features.

Primary emphasis is directed at minimizing the risk of fire by use of thermal insulation and adhesives which do not support combustion, flame retardant wiring, adequate overload and short circuit protection, and the elimination of combustible trim and furnishings. The facility is equipped with protection systems for controlling fires which might originate in plant equipment. See Sub-Chapter 9.8 for a description of the Fire Protection System.

The Containment and Auxiliary Building Ventilation Systems can be operated from the control room of the corresponding unit as required to limit the potential consequences of fire. Critical areas of the containment, the control room and the areas containing components of engineered safety features, have detectors to alert the control room to the possibility of fire so that prompt action may be taken to prevent significant damage.

Sharing of Systems

Criterion: Reactor facilities may share systems or components if it can be shown that such sharing will not result in undue risk to the health and safety of the public.

Two types of sharing were considered: a) sharing of systems and components between the two units and b) sharing of components among systems within a unit. For such shared systems and components, analyses confirm that there is no interference with basic function and operability of these systems due to sharing, and hence no undue risk to the health and safety of the public results. Sub-Chapter 1.3 identifies the shared facilities and equipment in the plant.

Missile Protection

Criterion: Adequate protection for those engineered safety features, the failure of which would result in undue risk to the health and safety of the public, shall be provided against dynamic effects and missiles that might result from plant equipment failures.

This section discusses in general terms the missile criteria, missile sources, and methods of missile protection for the Donald C. Cook Nuclear Plant.

A more comprehensive discussion of missiles arising in the event of a failure of the main turbine-generator can be found in Chapter 14.

Missile Protection Criteria

The Donald C. Cook Nuclear Plant is designed so that missiles from external or internal sources:

1. Will not cause or increase the severity of a loss of coolant accident.

2. Will not damage Engineered Safety Features such that the minimum required safety functions are jeopardized.

3. Will not cause a break in the Class I portion of a steam or feedwater pipe.

4. Will not prevent safe shutdown and isolation of the reactor plant.

5. Will not damage fuel stored in the Spent Fuel Pit.

Potential Missiles

Credible missiles, from sources considered capable of generating potential missiles, are defined as follows:

1. Tornadoes

   a. Bolted Wood Decking - 12 ft x 12 ft x 4 in, 450 lbs. traveling at 200 mph.

   b. Corrugated Sheet Siding - 4 ft x 4 ft, 100 lbs. traveling at 225 mph.

   c. Passenger car - 4000 lbs traveling along the ground at 50 mph.

   d. Small diameter pipe - 2 1/2 in., schedule 40, steel pipe 8 ft. length.

2. Main Turbine Failure

   General Electric Unit 1

   a. Vane from last stage bucket - 54 lbs traveling at 1170 ft per sec (casing exit velocity).

   b. 120° segment of last stage Wheel - 8264 lbs traveling at 409 ft per sec (casing exit velocity).

Brown Boveri Unit 2

a. Vane from last stage bucket - 168 lbs traveling at 1135 ft per sec (casing exit velocity).

b. 120° segment of next-to-last disc - 8360 lbs traveling at 551 ft per sec (casing exit velocity).

3. Structures and overhead cranes which are not of Seismic Class I design.

4. Dynamic equipment failures encompassing pumps, diesel engines, and turbine drives.

5. Valve stems and bonnets of significant size, having the potential to violate any of the missile protection criteria.

6. Control rod drive mechanism or parts thereof.

7. Pipe rupture whip, including steam/water jet forces following a pipe rupture of an adjacent pipe.

8. Miscellaneous.

   a. Sand plugs.
   b. Instrument wells and thimbles with mounted components.

With reference to item 7, above, to determine the dynamic impact and erosive effects of high temperature pressurized water and of steam jets from ruptured pipe lines, Westinghouse conducted a series of tests with subcooled water at 2250 psia/500°F and with saturated steam at 1030 psia, released through nozzles of 3 different diameters, impinging on reinforced concrete structures, at various angles. Evaluation of the results[2] indicates that erosion of concrete by a primary coolant or steam line break definitely does not impose a design consideration.

Missile Protection Methods

Protection of safety-related equipment from missiles has been accomplished by one or more of the following methods:

1. Compartmentalization

   Enclosing equipment in missile protected compartments.

2. Barriers

   Erecting barriers to stop potential missiles either at the source or at the location of the equipment to be protected.

3. Separation

   Sufficient separation of redundant systems so that a potential missile cannot impair both systems.

4. Restraints

   Limiting generation of potential missiles by means of restraints.

5. Equipment Design

   Designing the structure or component to withstand a missile, without loss of function.

6. Strategic Orientation

Orienting equipment, or parts of equipment, in a direction that directs the potential missile paths away from safety-related equipment.

7. Distance

Locating equipment beyond range of potential missiles.

Determination of Missile Shield Thickness

In cases where concrete or steel is used as missile protection, the calculation of the missile shield thickness required was based on the modified Petry formula, as set forth in the U. S. Navy Bureau of Yards and Docks publication, "Design of Protective Structures", Navy Docks P-51, or the Stanford Steel Penetration formula presented in Nuclear Engineering and Design, "The Design of Barricades for Hazardous Pressure Systems", C. V. Moore, 1967.

Records Requirements

Criterion: The reactor licensee shall be responsible for assuring the maintenance, throughout the life of the reactor, of records of the design, fabrication, and construction of major components of the plant essential to avoid undue risk to the health and safety of the public.

The Indiana and Michigan Electric Company or its authorized representative and Westinghouse Electric Corporation have retained documentation of the design, fabrication and construction of essential plant components.

These records verify the high quality and performance standards applicable to essential plant components.

1.4.2 PROTECTION BY MULTIPLE FISSION PRODUCT BARRIERS

Physical barriers are provided by the fuel pellet, fuel cladding, reactor coolant system pressure boundary and containment structure to protect the public from the release of fission products produced within the fuel assemblies. The specific details and design basis for each barrier are identified and discussed in Chapters 3, 4, and 5.

The design of the fuel cladding, core related structural equipment, and control and protective systems ensures that fuel damage in excess of acceptable limits is not likely, or can be readily suppressed in the unlikely event of its occurrence.

The Reactor Coolant System, including the reactor pressure vessel, was designed to accommodate the system pressure and temperatures attained under expected modes of plant operation, and to maintain material stress within applicable code stress limits. Its materials of construction are protected by control of coolant chemistry from corrosion phenomena. It is protected from overpressure by means of relieving devices.

High-pressure equipment in the Reactor Coolant System is surrounded by barriers to prevent a missile, generated from the Reactor Coolant system in a loss-of-coolant accident, from reaching either the containment liner or the containment cooling equipment, and from impairing the function of the engineered safety features. The principal missile barriers are the reinforced concrete operating floor and the reinforced concrete shield wall enclosing the reactor coolant loops. A steel and concrete structure was also provided over the control rod drive mechanisms to block a missile generated from a fracture of the mechanism housing.

The reactor coolant system piping and reactor vessel are completely enclosed within the containment structure. The containment structure itself was designed to withstand the temperature and pressure conditions associated with the complete severance of a reactor coolant pipe coincident with a seismic occurrence. Essentially no leakage of radioactive materials to the environment will result under these conditions.

1.4.3 NUCLEAR AND RADIATION CONTROLS

Monitoring potentially radioactive areas and operation of the reactor protection, reactor control systems and turbine-generator is accomplished in the control room from where actions required to maintain the safe operational status of the plant are centered.

Radiation protection has been provided to permit access to equipment in the control room, even under accident conditions, as necessary, to shut down and maintain safe control of the facility without radiation exposures to personnel in excess of the Code of Federal Regulations limits. The control room is equipped with the controls necessary for monitoring and maintaining control over the fission process and for conditions that could reasonably be expected to cause variations in core reactivity. In addition to instrumentation and controls which are required to maintain plant variables within prescribed operating ranges, means are provided to monitor fuel and waste storage handling areas, reactor coolant pressure boundary leakage, containment atmosphere and potentially contaminated facility effluent discharge paths.

Core protection systems automatically sense accident situations and initiate operation of the safety systems that prevent or suppress conditions that could result in exceeding fuel damage limits. This combination of monitoring and core protection systems provides assurance that radioactive releases are maintained well below established federal regulatory limits for normal operations, anticipated transients and possible accident conditions.

Positive indications in the control room of leakage of coolant from the reactor coolant system to the containment are provided by equipment which permits continuous monitoring of the containment air activity and humidity. The basic design criterion is the detection of deviations from normal containment environmental conditions including air particulate activity, radiogas activity, humidity, and in the case of gross leakage, the liquid inventory in the process systems and containment sump.

The containment atmosphere, unit vents, gland steam condenser vent, the condenser steam jet air ejector exhaust, steam generator power operated reliefs, and the Waste Disposal System liquid effluent are monitored for radioactivity.

For the case of leakage from the reactor containment under accident conditions, the plant area radiation monitoring system supplemented by portable survey equipment, provides adequate monitoring of releases during an accident.

Monitoring and alarm instrumentation have been provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels. Radiation monitors have been provided to maintain surveillance over the release of radioactive gases and liquids.

A controlled ventilation system removes gaseous radioactivity from the atmosphere of the spent fuel storage pool and waste treatment areas of the auxiliary building and discharges it to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate high-activity alarms on the control board annunciator, as described in Chapter 11.

1.4.4 RELIABILITY AND TESTABILITY OF PROTECTION SYSTEMS

Protection 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 this application, 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 rod drive mechanisms. The breaker main contacts are connected in series. Opening either breaker will interrupt power to all mechanisms causing all rods to fall by gravity into the core. Manual trip also actuates the shunt trip coil of the trip breakers. Each protection channel feeds two logic matrices, one for each undervoltage trip circuit.

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

The components of the protection 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, e.g., emergency core cooling pumps and containment spray systems, 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.

A comprehensive program of plant testing is executed for equipment vital to the functioning of engineered safety systems. The program consists of performance tests of individual pieces of equipment, and integrated tests of the system 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 can be conducted to assure that the systems can perform their design functions should they be called on during the plant lifetime.

a) Integrated Test Actuation Circuits and Motor-Operated Valves

The automatic actuation circuitry, valves and pump breakers can be checked during integrated system tests performed during each planned cooldown of the Reactor Coolant System for refueling.

b) Accumulator Tanks

The accumulator tank pressure and level are continuously monitored during plant operation and flow from the tanks can be checked at any time using test lines.

c) Safety Injection, Residual Heat Removal, Containment Spray and Centrifugal Charging Pumps

The centrifugal charging, safety injection, residual heat removal and containment spray pumps are periodically tested during plant operation in accordance with the applicable edition of the ASME Boiler and Pressure Vessel Code Section XI. Remotely operated valves in these systems are tested periodically within the criteria of ASME Section XI including exclusions and accepted code relief requests. Actuation circuits are tested periodically during plant operation or during plant shutdowns.

d) Boric Acid Concentration in the Accumulators

The accumulators are supplied with borated water at refueling water concentration of at least 2000 ppm while the plant is in operation. This concentration is checked periodically by sampling.

e) Boron Injection Tank

The Boron in this tank is maintained at a concentration of approximately 12 wt% boric acid (20,000-22,500 ppm boron).

f) Chemical Concentration in the Spray Additive Tank

The concentration of chemical solution in this tank is maintained at approximately 30 wt% NaOH.

g) Emergency Power Sources

The starting of the diesel-generator sets can be tested from the control room. The ability of the units to start within the prescribed time and to carry intended loads is checked.

h) Containment Penetration and Weld Channel Pressurization

Penetrations are designed with double seals and containment liner welds are backed by a steel channel. The large access openings such as the equipment hatch and personnel air locks are equipped with double gasket seals, and provisions are made for testing.

i) Instrumented Protection Channels

All reactor protection channels, with the exception of back-up reactor trips, are supplied in sets which provide the capability for channel calibration and test. Bypass removal of a trip circuit is used only in 2/4 logic which then becomes 2/3 logic, except for special 1/2 logic such as start-up trips which become 1/1 logic.

Reactor protection system protection channels in service at power are capable of being tested to verify operation. This includes a checking through to the final relay which forms the logic. Thus, the operability of a reactor trip channel can be determined conveniently and without ambiguity. A complete channel test can be performed through and including the final trip breakers, excluding the transmitter.

Actuation of the engineered safety features including containment isolation also employs coincidence circuits which allow checking of the operability of one channel at a time. Removal or bypass of one signal channel places that circuit in the tripped mode.

The normal on-line test procedure (exceptions noted above) consists of tripping the channel downstream of the on-off controller (process control) or superimposing the test signal on the transmitted signal (NIS Power Range). In the process control equipment, the 2/4 logic goes to 1/3 remaining, and the 2/3 logic goes to 1/2 remaining. The transmitted signal is disconnected and a simulated signal is injected. The trip points are then checked against this signal.

In the NIS power range equipment, a signal is superimposed on the existing input signal and the trip point is checked against the combined signal.

Transmitters and detectors are checked by comparing their outputs to each other.

1.4.5 REACTIVITY CONTROL

Two independent reactivity control systems, of different design principles, are provided in the reactor system design. These are neutron absorbing control rods and chemical poisoning of the reactor coolant with boron. The reactivity worth of the highest worth control rod is less than that required to achieve criticality with that rod out of the core and all the remaining control rods fully inserted in the core.

1.4.6 REACTOR COOLANT PRESSURE BOUNDARY

The Reactor Coolant System has been designed so that static and dynamic loads imposed on boundary components as a result of any inadvertent and sudden release of energy to the coolant will not cause rupture of the pressure boundary. In order to continually guard against any weakness developing, the reactor coolant pressure containing components have provisions for inspection and testing to assess the structural and leak-tight integrity of the boundary components during their service lifetime.

1.4.7 ENGINEERED SAFETY FEATURES

The engineered safety features provided in this plant have sufficient redundancy of components and power sources so that under the conditions of the design basis accident, the system can, even when operating with partial effectiveness, maintain the required integrity of fission product barriers to keep exposure of the public well within the guidelines of 10 CFR 100.

A general explanation of each of the engineered safety features is given below. Specific details on system design and operation are covered in Chapter 6.

1. A steel lined concrete containment structure provides an extremely reliable final barrier against the escape of fission products.

2. An emergency core cooling system is provided to deliver borated water to the core, in the event of a loss-of-coolant accident, in three modes: passive accumulator injection, active safety injection, and residual heat removal recirculation. The 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 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 limits 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, transferred to the refueling cavity, and placed in the fuel transfer system. 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. 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 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 Sub-Chapter 14.2 to result in exposures well within the limits of 10 CFR 100.

The refueling cavity, the refueling canal, the 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 transfer tube which connects the refueling canal and the transfer canal which forms part of the reactor containment is provided with a valve and a blind flange which closes off the 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.

TABLE 1.4-1

INDEX OF AEC GENERAL DESIGN CRITERIA

AEC Criterion Number	Criterion Title	FSAR Chapters
1	Quality Standards	1, 4, 5
2	Performance Standards	1, 4, 5, 8
3	Fire Protection	1, 5, 7
4	Sharing of Systems	1, 6, 9
5	Records Requirements	1, 4
6	Reactor Core Design	3
7	Suppression of Power Oscillations	3
8	Monitoring Reactor Coolant Leakage	4
9	Reactor Coolant Pressure Boundary	4
10	Reactor Containment	5
11	Control Room	7
12	Instrumentation and Control Systems	7
13	Fission Process Monitors and Control	7
14	Core Protection Systems	7
15	Engineered Safety Features Protection Systems	7
16	Monitoring Reactor Coolant Pressure Boundary	4
17	Monitoring Radioactivity Releases	11
18	Monitoring Fuel and Waste Storage	11
19	Protection Systems Reliability	7
20	Protection Systems Redundancy and Independence	7
21	Single Failure Definition	1
22	Separation of Protection and Control Instrumentation Systems	1
23	Protection Against Multiple Disability for Protection Systems	7
24	Emergency Power for Protection Systems	8, 10

TABLE 1.4-1 (cont'd.)

AEC Criterion Number	Criterion Title	FSAR Chapters
25	Demonstration of Functional Operability of Protection Systems	7
26	Protection Systems Fail - Safe Design	7
27	Redundancy of Reactivity Control	3, 7, 9
28	Reactivity Hot Shutdown Capability	3, 9
29	Reactivity Shutdown Capability	3, 9
30	Reactivity Hotdown Capability	3, 9
31	Reactivity Control Systems Malfunction	3, 7, 9
32	Maximum Reactivity Worth of Control Rods	3
33	Reactor Coolant Pressure Boundary Capability	4
34	Reactor Coolant Pressure Boundary Rapid Propagation Failure Prevention	4
35	Reactor Coolant Pressure Boundary Brittle Fracture Prevention	4
36	Reactor Coolant Pressure Boundary Surveillance	4
37	Engineered Safety Features Basis for Design	5, 6
38	Reliability and Testability of Engineered Safety Features	5, 6
39	Emergency Power	8
40	Missile Protection	1, 4, 5, 6, 7, 8, 9
41	Engineered Safety Features Performance Capability	5, 6, 9
42	Engineered Safety Features Components Capability	6
43	Accident Aggravation Prevention	6
44	Emergency Core Cooling System Capability	6
45	Inspection of Emergency Core Cooling System	6

from experimental and analytical development programs into the core thermal design codes used to evaluate the loss-of-coolant accident.

This program has been completed. A preliminary evaluation of the loss-of-coolant accident utilizing the results of the Flashing Heat Transfer Program in the core thermal design code has been presented in Reference 18.

8. <u>Blowdown Forces Program (Item 15 in Reference 1)</u>

The objective of the program was to develop digital computer programs for the calculation of pressure, velocity, and force transients in the Reactor core and internals during a loss-of-coolant accident, and to utilize these codes in the calculation of blowdown forces on the fuel assemblies and reactor internals to assure that the stress and deflection criteria used in the design of these components are met.

Westinghouse has completed the development of BLODWN-2, an improved digital computer program for the calculation of local fluid pressure, flow and density transients in the Reactor Coolant System.

Extensive comparisons have been made between BLODWN-2 and available test data, and the results are given in Reference 19. Agreement between code predictions and data has been good.

An analysis using the BLODWN-2 Program has been applied to this plant. It was concluded from the analysis that the design of this reactor meets the established design criteria.

9. <u>Gross Failed Fuel Detector Program</u>

Since the Donald C. Cook Nuclear Plant will not use the <u>W</u> delay neutron failed fuel monitor, the <u>W</u> R & D on this monitor is no longer applicable.

A description of the Failed Fuel Detection System to be used at the Donald C. Cook Nuclear Plant is given.

10. Reactor Vessel Thermal Shock (Item 16 in Reference 1)

The effects of safety injection water on the integrity of the reactor vessel following a postulated loss-of-coolant accident, have been analyzed using data on fracture toughness of heavy section steel both at beginning of plant life and after irradiation corresponding to approximately 40 years of equivalent plant life. The results show that under the postulated accident conditions, the integrity of the reactor vessel is maintained.

Fracture toughness data is obtained from a Westinghouse experimental program which is associated with the Heavy Section Steel Technology (HSST) Program at ORNL and Euratom programs. Since results of the analyses are dependent on the fracture toughness of irradiated steel, efforts are continuing to obtain additional fracture toughness data. Data on two-inch thick specimens is expected in 1970 from the HSST Program. The HSST is scheduled for completion by 1973.

A detailed analysis considering the linear elastic fracture mechanism method, along with various sensitivity studies was submitted to the AEC Staff and members of the ACRS enlisted: "The Effects of Safety Injection On A Reactor Vessel And Its Internals Following A Loss Of Coolant Accident" (December, 1967), (Proprietary). Revised material for this report plus additional analysis and fracture toughness data was presented at a meeting with the Containment and Component Technology Branch on August 9, 1968, and forwarded by letter for AEC review and comment on October 29, 1968.

TABLE 1.6-1 (cont'd)

198. NUREG/CR 3988, "MARCH-2, Meltdown Accident Response Characteristic Code Description and Users Manual", BMI-2115, Battelle Columbus Laboratories, September 1984, R. O. Wooten, et. al.

199. NUREG-75/057, "TOODEE2: A Two-Dimensional Time dependent Fuel Element Thermal Analysis Program," May 1975, G. N. Lauben.

200. XN-76-51, Supplement 1, "Flow Blockage and Exposure Sensitivity Study for D. C. Cook Unit 1 Reload Fuel Using ENC WREM-II Model," January 1977, K. P. Galbraith et. al.

201. XN-76-51, Supplement 2, "Flow Blockage and Exposure Sensitivity Study for ENC D. C. Cook Unit 1 Reload Fuel Using ENC WREM-2 Model," January 1978, G. C. Cooke.

202. XN-76-51, Supplement 3, "Flow Blockage and Exposure Sensitivity Study for ENC D. C. Cook Unit 1 Reload Fuel Using ENC WREM-2 Model," March 1978, R. E. Collingham et. al.

203. XN-NF-78-30, Amendment 1, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-IIA: Response to NRC Request for Additional Information," February 1979, S. E. Jensen et. al.

204. XN-NF-81-07, "LOCA ECCS Reanalysis for D. C. Cook Unit 1 Using the ENC WREM-IIA PWR ECCS Evaluation Model," February 1981, S. E. Jensen et. al.

205. XN-NF-81-58(P), Revision 2, "RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model," January, 1983, K. R. Merckx, Ed.

206. XN-NF-82-07(P), "Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model," March 1982, W. V. Kayser.

207. XN-NF-82-20(P), Supplement 2, "Exxon Nuclear Company Evaluation Model EXEM/PWR ECCS Model Updates: Large Break Example Problem for 4-Loop PWR with Ice Condenser," March 1982, T. Tahvili.

AMERICAN ELECTRIC POWER Company, Inc.

AEP

1 Riverside Plaza (614) 223-1000
P.O. Box 16631
Columbus, Ohio 43216-6631

W. S. WHITE, JR.
Chairman of the Board
and
Chief Executive Officer
(614) 223-1500

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 Electric Company's Donald C. Cook Nuclear Plant (Cook 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 for safety-related activities performed during the operations of, or in support of the Cook Plant. The Quality Assurance Program supports the goals of maintaining the safety and reliability of the Cook Plant at the highest level, and conducting safety-related activities in compliance with 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 Plant. I have delegated functional responsibility for the Quality Assurance Program to the American Electric Power Service Corporation (AEPSC) Vice Chairman - Engineering and Construction. He has, with my approval, delegated further responsibilities as outlined in this statement.

IMPLEMENTATION

The AEPSC Manager of Quality Assurance, under the direction of the AEPSC Vice Chairman - Engineering and Construction, has been assigned the overall responsibility for specifying the Quality Assurance Program requirements for the Cook Plant and verifying their implementation. The AEPSC Vice Chairman - Engineering and Construction has given the AEPSC Manager of Quality Assurance authority to stop work on any quality-related activity that does not meet applicable administrative, technical and/or regulatory requirements. The AEPSC Manager of 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 Vice President - Nuclear Operations and the Executive Vice President and Chief Engineer, under the direction of the AEPSC Vice Chairman - Engineering and Construction, have been delegated responsibility for effectively implementing the Quality Assurance Program.

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

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

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

Each AEPSC and Cook Plant organization that is, or becomes, involved in safety-related activities for the Cook Plant has the responsibility to implement the policies and requirements of the Quality Assurance Program that are applicable to their respective area(s) of responsibility. AEPSC and Cook Plant personnel involved in safety-related activities shall be familiar with, and comply with, the requirements of the applicable Quality Assurance Program requirements.

COMPLIANCE

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

Any employee engaged in safety-related activities who believes that the Quality Assurance Program is not being complied with, or that a deficiency in quality exists, should notify his or her supervisor, the AEPSC Manager of Quality Assurance and/or the Plant Manager. If the notification does not in the employee's opinion receive prompt 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.

W. S. White, Jr.
Chairman of the Board
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 Program for the operational phase of the D.C. Cook Nuclear Plant (Cook 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 and annually thereafter as a minimum.

This section of the Quality Assurance Program Description 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 Quality Assurance Program, and verifying that activities affecting the quality of safety-related items are performed in accordance with QA Program requirements.

1.7.1.2 IMPLEMENTATION

1.7.1.2.1 <u>Source of Authority</u>

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 Donald C. Cook Nuclear Plant. Authority and responsibility for effectively implementing the QA Program for plant modifications, operations and maintenance are delegated through the AEPSC Vice Chairman - Engineering and Construction, to the AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations) and the AEPSC Executive Vice President and Chief Engineer (reference John E. Dolan letter dated November 1, 1984, Subject: Support Organization for Donald C. Cook Nuclear Plant).

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. While the AEPSC organization changes effective on September 1, 1984, have not affected the responsibility and authority of the Manager of Nuclear Operations, these changes in the AEPSC engineering organization require a new directive for the support of the Cook Plant.

The AEPSC corporate support of the Cook Plant is the responsibility of the entire organization under the direction of the Manager of Nuclear Operations who maintains primary responsibility for the Cook 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 Department, having a supporting role for the Cook 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 nuclear plant operations and support.

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

3) AEPSC division and department managers responsible for nuclear 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 the Cook Plant.

4) AEPSC division and department managers responsible for nuclear 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 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 division and department managers having nuclear support responsibilities as well as the Plant Manager and plant department 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 and the referenced subject letter.

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

The American Electric Power Company, Inc., (AEP) Chairman of the Board and Chief Executive Officer has delegated the functional responsibility of the Quality Assurance Program to the American Electric Power Service Corporation (AEPSC) Vice Chairman - Engineering and Construction.

The AEPSC Manager of Quality Assurance, under the direction of the AEPSC Vice Chairman - Engineering and Construction, is responsible for specifying Quality Assurance Program requirements and verifying their implementation.

The AEPSC Vice President - Nuclear Operations and AEPSC Executive Vice President and Chief Engineer, under the direction of the AEPSC Vice Chairman - Engineering and Construction, are responsible for effectively implementing the Quality Assurance Program.

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

Management/supervisory personnel receive functional training to the level necessary to plan, coordinate, and administrate those day-to-day verification activities of the QA Program for which they are responsible.

AEPSC has established 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 Plant. The function of the NSDRC is to oversee the engineering, design, operation, and maintenance of the Cook Plant by performing audits and independent reviews of activities which are specified in the Facility Operating Licenses.

The Cook Plant on-site review group is the Indiana & Michigan Electric Company (I&MECo) Plant Nuclear Safety Review Committee (PNSRC). This committee has been established pursuant to the requirements of the Cook 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 of AEP is the Chief Executive Officer of 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 and Chief Executive Officer.

American Electric Power Service Corporation

The responsibility for administrative and technical direction of the AEP System and its facilities is delegated to the American Electric Power Service Corporation (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 function responsibility is assigned to the operating companies. In the case of Cook Plant, I&MECo provides only public affairs, accounting and industrial safety direction.

The Donald C. Cook Nuclear Plant is owned and operated by Indiana & Michigan Electric Company (I&MECo) which is part of the AEP system.

1.7.1.2.4 Quality Assurance Responsibility of AEPSC

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

2) AEPSC furnishes licensing, NRC correspondence, fuel management and radiological support activities.

3) AEPSC provides additional service in matters such as supplier qualification, and spare and replacement part procurement, to the extent established by AEPSC and plant procedures.

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

5) Cognizant Engineer - The AEPSC engineer that provides overall engineering and design responsibility, including implementation of quality assurance and quality control measures, for a system, item of equipment, or structure.

Quality Assurance Responsibility of I&MECo - D.C. Cook Plant

As owner and operator, I&MECo operates the Cook Plant per 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 Plant.

1.7.1.2.5 Organization (AEPSC)

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

- AEPSC Manager of Quality Assurance
- AEPSC Vice President - Nuclear Operations
- AEPSC Executive Vice President and Chief Engineer

Quality Assurance Department

The AEPSC Manager of Quality Assurance reports to the AEPSC Vice Chairman - Engineering and Construction and is responsible for the Quality Assurance Department. The Quality Assurance Department consists of the following positions and sections (Figure 1.7-4):

- Quality Assurance Engineering Section
- Audits and Procurement Section
- Training and Procedures Specialist
- Quality Assurance Staff Specialist
- D.C. Cook Plant Site Quality Assurance Section

The Quality Assurance Department is organizationally independent and is responsible to perform the following:

- Identify quality problems.
- Initiate, recommend, or provide solutions through designated channels.
- Verify implementation of solutions.
- Prepare issue and maintain Quality Assurance Program documents, as required.
- Verify the implementation of the Quality Assurance Program through scheduled audits and surveillances.
- Review 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 orientation, training, certification and qualification of AEPSC personnel.
- Provide general guidance, when requested, for the collection, storage, maintenance, and retention of quality assurance records.
- Establish and maintain a Qualified Suppliers List (QSL) of nuclear (N) items and services.

- 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.
- Assure that conditions adverse to quality are dispositioned to preclude recurrence.
- Conduct in-process QA surveillance at supplier's facilities, as required.
- Assist and advise other AEP/AEPSC groups in matters related to the Quality Assurance Program.
- Maintain a list of nuclear grade items (N-List) for the D.C. Cook Plant.
- Establish a mechanism for identifying, tracking and closing out quality-related commitments.
- Conduct audits as directed by the Nuclear Safety and Design Review Committee (NSDRC).
- Review AEPSC originated nonconformances, noncompliances and associated corrective action recommendations.
- Maintain cognizance of industry and governmental quality assurance requirements such that the Quality Assurance Program is compatible with requirements, as necessary.
- Recommend for revision to, or improvements in the established QA Program to senior AEPSC management.
- Issue "Stop Work" orders when significant conditions adverse to quality 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.
- Routine attendance and participation 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.

Amplification of Specific Responsibilities

- Qualification of the AEPSC Manager of Quality Assurance

  The AEPSC Manager of Quality Assurance shall possess the following position requirements:

  - Bachelor's degree in engineering, scientific or related discipline.
  - Ten (10) years experience in one or a combination of the following areas: engineering, design, construction, operations, maintenance of fossil or nuclear power generation facilities or utility facilities Quality Assurance, 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 quality 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.

- Stop Work Orders

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

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

- <u>QA Orientation, Training, Qualification and Certification Program</u>

  a) AEPSC QA shall, if directed by AEPSC management, be responsible for establishing, maintaining and conducting a general QA orientation and training program for AEPSC personnel engaged in safety-related activities. This program includes the AEPSC QA philosophy and such facility specific programs as may be required by facility or regulatory requirements.

  b) AEPSC has established and maintains a QA Auditor training and certification program for all AEPSC QA Auditors.

- <u>Problem Identification, Reporting and Escalation</u>

  - AEPSC QA has established mechanisms for the identification and reporting and escalating safety-related problems to a level of management whereby satisfactory resolutions can be obtained.

<u>Nuclear Operations Division</u>

The AEPSC Vice President - Nuclear Operations (Manager of Nuclear Operations) reports to the AEPSC Vice Chairman - Engineering and Construction and is responsible for the Nuclear Operations Division. Reporting to the AEPSC Vice President - Nuclear Operations are the following:

- Donald C. Cook Plant Manager
- Assistant Division Manager - Nuclear Engineering (not charted)
- Assistant Division Manager - Nuclear Operations (not charted)
- Consulting Nuclear Engineer - Nuclear Operations (not charted)
- Staff Engineer - Nuclear Operations (not charted).

The organization and responsibilities of the Donald C. Cook Plant Manager are defined further within this section under 1.7.1.2.6 Organization (Cook Plant).

The AEPSC Assistant Division Manager - Nuclear Engineering is responsible for two of the four sections within the Nuclear Operations Division, as follows (not charted):

- Nuclear Safety and Licensing (NS&L) Section
- Nuclear Material and Fuels Management (NMFM) Section

The AEPSC Assistant Division Manager - Nuclear Operations is responsible for the remaining two sections, as follows (not charted):

- Nuclear Operations Support (NOS) Section
- Radiological Support (RS) Section

The Nuclear Operations Division 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.
- Provide direction to all other AEPSC engineering divisions on engineering matters pertaining to the Cook Plant.
- Maintain liaison with the AEPSC Manager of Quality Assurance.
- Implement the requirements of the AEPSC Quality Assurance Program.
- Maintain knowledge of the latest safety, licensing, and regulatory requirements, codes, standards and federal regulations applicable to the operation of Cook 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, to evaluate bids, and to negotiate and administer contracts for the procurement of all nuclear fuel and related components and services.
- Prepare testimony for, and participate in Public Service Commission proceedings concerning nuclear fuel costs and related rates charged to the customer.
- Keep special nuclear material accountability records.
- 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 in these activities.
- Develop, maintain and implement a quality assurance program both for the specific fabrication of nuclear fuel and related components and for auditing the quality program of the vendors of these products.
- Contract for, and provide technical support for, disposal of both high level and low level radioactive waste.
- Obtaining and maintaining the NRC Operating License and Technical Specifications for the Cook 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, prepare criteria, reports, and responses, reviewing items affecting safety, and interpreting regulations.
- Review, coordinate, and resolve all matters pertaining to nuclear safety between Cook Plant and AEPSC. This includes, but is not limited to: the review of certain plant modifications to ensure that the requirements of 10CFR50.59 are met; the preparation of safety evaluations or reviews for any designated subject; 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.

- Provide the corporate cognizant safety engineer who is responsible for all matters associated with nuclear safety.
- Primary corporate contact for most oral and written communication with the NRC.
- Corporate representative to the Westinghouse Owners Group.
- Provide the support in key areas of expertise such as nuclear engineering, probabilistic analysis, thermohydraulic analysis, chemical engineering, mechanical engineering, electrical engineering, and technical writing.
- Provide the secretary of the Nuclear Safety and Design Review Committee and coordinate and report on committee meetings.
- 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 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 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 (Request for Changes), to the extent that they are related to the ultimate safe operation of the plant, for compliance with safety regulations, plant Technical Specifications, the FSAR design basis, and with any other requirements

under the Operating License and to determine if there are any unreviewed safety questions as defined in 10CFR50.59.

- Perform reviews of Noncompliance Reports and 10CFR21 reviews in accordance with corporate requirements.
- Provide as a focal point within AEPSC for coordinating design changes for the Cook Plant. 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 Plant. These responsibilities for both capitalized and expensed modifications and additions to Cook Plant.
- Provides working level coordination with the INPO. This effort includes providing AEPSC access to INPO resources such as NUCLEAR NETWORK and NRPDS, and effectively integrating AEPSC and Cook Plant efforts towards utilizing INPO recommendations contained in Operating Experience Reports to improve Cook Plant performance.
- Coordinate the AEPSC review of completed plant condition reports and provide organizational services and record keeping for review work performed by the NSDRC Subcommittee on Corporate and Plant Occurrences.
- Coordinate AEPSC inputs for Cook Plant operating and maintenance budgets, review these budgets, present the budgets to AEPSC management, and monitor and assess budget performance.
- Daily communication with the Cook Plant, provide AEPSC management with a daily plant status report, and makes presentations to senior management at regularly scheduled construction staff meetings.
- Provide administrative coordination for the Ice Condenser Task Force and for the Regulatory Performance Improvement Program (RPIP).
- Obtaining a plant simulator and developing master service contracts.
- Process incoming vendor information.
- Coordinate development of a plant facility data base.
- Participate in human factors reviews, and contributing to the annual FSAR updates through reviews of Licensee Event Reports and the Annual Operating Report.
- Radiological, emergency and security planning.

- Corporate support of the Cook Plant's radiation protection and health physics program, technical service and advice on the radiological aspects of design changes, modifications or capital improvements, the ALARA program, the radiation monitoring system, the environmental radiological monitoring and sampling program, dose and shielding analysis, radiochemistry review, and meteorological monitoring.
- Cook 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.
- 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 and radiation training for coal plant personnel who handle radiation sources.
- Participate on ALARA Subcommittees.
- 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.

Environmental Engineering Division

The AEPSC Executive Vice President and Chief Engineer, reporting to the AEPSC Vice Chairman - Engineering and Construction, is responsible for the Environmental Engineering Division through the AEPSC Assistant Vice President - Environmental Engineering. The Environmental Engineering Division provides a nonsafety-related function for the Cook Plant with exception of its participation on the Nuclear Safety and Design Review Committee (NSDRC).

Engineering and Design

The AEPSC Executive Vice President and Chief Engineer, reporting to the AEPSC Vice Chairman - Engineering and Construction, is responsible for certain engineering and design functions through the AEPSC Vice President - Engineering and Design. The AEPSC Vice President - Engineering and Design is responsible for the following divisions:

- Civil Engineering Division
- Design Division
- Materials Handling Division

Civil Engineering Division

The AEPSC Division Manager - Civil Engineering, reporting to the AEPSC Vice President - Engineering and Design, is responsible for the Civil Engineering Division. The Civil Engineering Division consists of the following (not charted):

- Structural Engineering Section
- Civil Engineering Laboratory Section
- Geotechnical Engineering Section
- Survey and Mapping Group

The Civil Engineering Division is responsible for the following:

- Make recommendations and assist in the formulation of policies and practices relating to the structural design and engineering of office and service buildings, and miscellaneous structures, and provide the general supervision of the structural engineering of such facilities and structures.
- Arrange for outside engineering and consulting assistance as required.
- Prepare and review improvement requisitions for capital expenditures.
- Approve invoices for outside services.
- Approve purchase requisitions and contracts as authorized.
- Prepare and approve Request for Changes (RFCs) pertaining to nuclear generating plants.
- Initiate and maintain a program of development and training for personnel in the division.

- Prepare specifications, procurement of civil/structural works and modifications to same relative to the Civil Engineering Division.
- Direct and coordinate the preparation of specifications and instructions to bidders for general construction and structural features of power plants and buildings and evaluate proposals received; make recommendations for the award of contracts.
- Direct and coordinate the preparation of contracts for the structural phases of power plant and building design and construction.
- Provide services to the field organizations, including the assignment of personnel to the field during construction, normal or emergency outages, or as requested.
- Assist in planning and execution of maintenance work on buildings and other structures.
- Prepare site studies.
- Arbitrate disputes which arise between construction forces and outside suppliers of materials and services.
- Coordinate structural consultant's reports with design.
- Participate in periodic inspections of contractors' work.
- Check of structural drawings submitted for review.
- Review and recommend concrete mix formulations for all new construction.
- Supervise maintenance and repairs of all masonry and concrete work in the AEP System, including supplying trained inspection personnel.
- Direct testing of materials used in concrete and testing of soils to be used in work throughout the AEP System.

Design Division

The AEPSC Division Manager - Design, reporting to the AEPSC Vice President - Engineering and Design, is responsible for the Design Division. There are two (2) Assistant Division Managers (not charted) reporting to the AEPSC Division Manager - Design who are responsible for various sections as follows (not charted):

Assistant Division Manager

- Architectural Design Section
- Mechanical Design Section
- Structural Design Section

Assistant Division Manager

- Electrical Plant Section
- Control Services Section

The Design Division is responsible for the following:

- Formulate, administer, and implement policies and practices relating to the design of power plants and miscellaneous structures.
- Direct the development, maintenance, procedural review and implementation by which the Design Division adheres to the QA Program elements as established by the AEPSC General Procedures Manual.
- Conduct periodic management reviews and surveillances of division activities to ensure compliance with QA Program objectives, and external surveillances as necessary, of consultants outside organizations and vendors for which the division is cognizant.
- Conduct functions of the division so as to be in conformance with the operating licenses of the Cook Plant.
- Coordinate the review and/or answering of corrective actions issued and assigned to the Design Division.
- Coordinate special projects and studies, as required.
- Establish and maintain files of design documents for record purposes.
- Initiate and/or implement and control design changes and modifications.
- 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 Plant including monitoring of the Aperture Card Microfilm Program.
- Supervise and control the work of consultants, Architect/Engineers and outside design agencies supplying services to AEP in their .

discipline and process notification of defects in accordance with company requirements. Also perform detailed reviews of design work submitted by outside agencies.

- Supervise the identification of critical design decisions and ensure appropriate analyses and reviews are provided. Review, approve and/or sign off all design drawings prior to issuance.
- Provide to the field organizations such services as required during construction, normal or emergency outages or as requested, including assigning design personnel to the field.
- Maintain an up-to-date list of all major approved materials and specifications used within the division's scope of responsibility.
- Initiate and/or aid in the responses of reportable items as described in the AEPSC General Procedures and division procedures.
- Schedule, develop, coordinate and control design studies calculations/analysis, drawings, purchase documents, specifications and other design activities, as assigned for system, components or structures within the division's responsibility.
- Review and update, as required, the Cook Plant Final Safety Analysis Report (FSAR).
- Perform functions related to the Cook Plant as required in response to NRC requirements.
- Participate on committees that review nuclear activities as appointed or assigned.
- Coordinate and resolve design comments made by interfacing departments/divisions.
- Prepare, review approve and administer design specifications and purchase documents for design services and/or materials.
- Initiate and/or aid in the responses of reportable items as described in the AEPSC General Procedures and division procedures.
- Participate in the Initial Assessment Group (IAG) and provide assistance to on-site personnel and other divisions.
- Identify and report deficiencies in the division's functions, duties, and responsibilities.
- Coordinate the implementation of division commitments.

Materials Handling Division

The AEPSC Division Manager - Materials Handling, reporting to the AEPSC Vice President - Engineering and Design, is responsible for the Materials Handling Division. The Materials Handling Division contains one (1) section that performs safety-related work as follows (not charted):

- Coal and Materials Handling Section

The Coal and Materials Handling Section is responsible for the following:

- Develop policies and practices relating to the engineering of materials handling installations for Donald C. Cook Nuclear Plant.
- Review the activities of materials handling systems for the Cook Plant and approve, as required, all design changes and modifications including the preparation of specifications, procurement of equipment and modifications to equipment.
- Arrange for outside engineering and consulting services, as required.
- Provide training and development programs necessary for personnel of the division (including the company's safety and health program), which are consistent with the written policy of American Electric Power Company and American Electric Power Service Corporation.
- Prepare and administer erection and service contracts.
- Review and evaluate proposals and make recommendations for awards of purchase orders and contracts.
- Prepare, review and approve specifications, purchase and change documents, sketches, drawings, design input, design verifications and calculations, as required.
- Initiate and/or review approval and control of laboratory and field investigations, feasibility studies, improvement requisitions, reports and cost estimates pertaining to the Cook Plant.
- Provide field services to the Cook Plant including the assigning of personnel as are required during construction, normal or emergency outages, or as requested.
- Direct the review of, and response to corrective actions assigned to the Material Handling Division.
- Identify critical engineering and design input and ensure that appropriate analysis and reviews are conducted.

- Implement a corrective action system with regard to all safety-related activities of the division that will control and document all items, services, or activities which do not conform to requirements.
- Maintain a surveillance program in support of the Quality Assurance Program and review and approve the activities of this program which can be separated into the following two (2) areas:
    - Internal management review of the Materials Handling Division.
    - External technical surveillance of consultants, outside materials handling organizations and vendors over which the division is cognizant.
- Assist in planning and execution of maintenance work on equipment and facilities.
- Review and approve manufacturer's equipment drawings prior to fabrication.
- Prepare design criteria, engineering standards, conceptual layouts, studies and procedures in conjunction with materials handling equipment at the Cook Plant.
- Assist in the preparation of applications for federal, state and local permits relative to installations being made which require such permits.
- Perform shop and field inspections on equipment being fabricated or installed which is within the scope of the division's responsibility.
- Provide input for special studies and reports which may be requested by other divisions or governmental agencies such as the Nuclear Regulatory Commission.
- Provide technical guidance when requested in support of maintenance and operations activities at the Cook Plant.
- Conduct periodic management reviews of the activities of the division to ensure compliance with the objectives of the Quality Assurance Program, and external technical surveillance, as necessary, of consultants, outside materials handling organizations and vendors over which the division is cognizant.
- Establish and maintain a permanent file for QA records.
- Process RFCs in accordance with AEPSC General Procedures and division procedures.

Electrical Engineering Department

The AEPSC Executive Vice President and Chief Engineer, reporting to the AEPSC Vice Chairman - Engineering and Construction, is responsible for the Electrical Engineering Department through the AEPSC Senior Vice President - Electrical Engineering and Deputy Chief Engineer. Reporting to AEPSC Senior Vice President - Electrical Engineering and Deputy Chief Engineer is the AEPSC Manager - Generation and Telecommunications Engineering Division. The Generation and Telecommunications Engineering Division (not charted) is the only division within the Electrical Engineering Department that is responsible for performance of electrical oriented safety-related activities. The AEPSC Assistant Manager -

Generation and Telecommunications Engineering Division reports to the AEPSC Manager - Generation and Telecommunications Engineering Division and is responsible for the one (1) section within the Electrical Engineering Department that is responsible for safety-related activities as follows (not charted):

- Electrical Generation Section

The Electrical Generation Section is responsible for the following:

- Plan and engineer, in conjunction with other specialists, sections and divisions, electrical facilities inside Cook Plant up to the high voltage (HV) bushings of the main generator transformers, and the relaying and controls on breakers associated with the generator and auxiliary system, including: determination of general layout and design; advising on selection of major electrical equipment; preparation of one-line diagrams, and; coordination of inside and outside electrical plant facilities.
- Engineer and design all electrical controls for operation and protection of steam generator, turbine generator, and auxiliary equipment and general plant protection, including checking elementary diagrams and approving drawings.

- Prepare cost estimates and improvement requisitions for electrical plant facilities, including review of improvement requisitions and cost estimates prepared by others.
- Review and approve all procedures, correspondence, requests for design changes or modifications as appropriate.
- Obtain, review and perform engineering evaluations including equipment qualification.
- Provide technical support to Nuclear Safety and Licensing (NS&L) and to Cook Plant Operations and Maintenance Departments.
- Perform and evaluate economic studies, investigations, analysis and reports for electrical facilities pertaining to the design, operation and maintenance of the generating plants.
- Maintain a constant awareness for improvements and more economic design of equipment, electric facilities, maintenance and operating methods or procedures.
- Assign membership to the Nuclear Safety and Design Review Committee (NSDRC) audit subcommittees, participating in matters covered in the committee's charter.
- Participate in the evaluation and remedy of any situation requiring activation of the emergency response organization.
- Prepare and/or approve specifications and purchase requisitions, and perform drawing review of electrical equipment, including control and protective relays.
- Assist field personnel in installation, start-up and the subsequent locating of problems in protective, control, or electrical equipment and in determining proper operation of equipment during normal or after emergency operations.
- Assist with the establishing of relay and control standards.
- Maintain a constant awareness of the activities of the ensure compliance with all applicable procedures initiating, when required, training or retraining programs.
- Review and approve responses to NRC correspondence as required.
- Closely follow manufacturers' engineering and designs to ensure provision of adequate and reliable equipment and circuitry in the areas of turbine-generator protective controls, switchgear, elec-

trical auxiliaries, mechanical equipment and protective devices upon which depend the safety, reliability, economy and performance of the unit and plant.

- Perform calculations for proper application and settings of protective relays.
- Coordinate with the Mechanical Engineering Division to ensure that all electrical devices purchased with mechanical equipment conform to accepted standards and fulfill the desired function.

Mechanical Engineering Division

The AEPSC Executive Vice President and Chief Engineer, reporting to the AEPSC Vice Chairman - Engineering and Construction, is responsible for the Mechanical Engineering Division through the AEPSC Assistant Vice President - Mechanical Engineering. Reporting to the AEPSC Assistant Vice President - Mechanical Engineering, are the following (not charted):

- AEPSC Assistant Division Manager(s)
- Consulting Mechanical Engineer - Nuclear
- Staff Engineer - Chief Metallurgist

Further, the AEPSC Assistant Division Manager - Nuclear is responsible for the following positions and sections (not charted):

- Nuclear Project Engineer(s)
- Turbine and Cycle Evaluation Section
- Chemical Engineering Section
- Heat Exchangers and Pumps Section
- Piping and Valves Section
- Instrumentation and Control Section
- Fire Protection and HVAC Section
- Analytical and R&D Section

The Mechanical Engineering Division is responsible for the following:

- Provide technical engineering support in areas of operation and maintenance, including: the Inservice Inspection (ISI) Program; the Quality Assurance Program; the AEP ALARA Program covering radiation protection, and; the corporate and plant Industrial Safety Program.

- Provide engineering support for the other AEPSC engineering divisions, as well as for the manufacturers, suppliers, or constructors of equipment and systems.
- Provide engineering support to the AEPSC Nuclear Operations Division.
- Preparation of equipment specifications and purchase requisitions for plant equipment, major spare parts and services related to specific areas of responsibility of MED.
- Provide technical direction and assistance to the AEPSC Design Division in the layout and arrangement of equipment, piping, systems, controls, etc., for the development of drawings.
- Develop system flow diagrams and progressive reviews to determine the adequacy of system designs.
- Provide technical assistance to the Cook Plant for use and control of special processes, including welding, heat treating, nondestructive examination, etc.
- Initiate and develop design changes in areas of responsibility of the Mechanical Engineering Division.
- Develop System Descriptions and Descriptive Articles.
- Provide support personnel for the emergency response organization.
- Provide analytical support in engineering disciplines (e.g., heat transfer, thermodynamics, fluid dynamics).
- Review and approval of mechanical design drawings.
- Provide Engineering evaluations for Condition Reports, LERs, INPO-SOERs and NRC Bulletins.

Plant Construction Division

The AEPSC Assistant Vice President - Plant Construction Division reports to the AEPSC Vice Chairman - Engineering and Construction, and is responsible for the Plant Construction Division. The Plant Construction Division consists of the following sections (not charted):

- Administrative Section
- Construction Contracts Section

The Plant Construction Division is responsible for the following:

- Provide a Construction Manager, reporting administratively to the AEPSC Assistant Vice President - Plant Construction Division and functionally to the Cook Plant Manager, to perform major modifications and maintenance work.
- Scope, bid and make recommendations relative to construction contracts.
- Administer contracts throughout the construction period.

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 Stores.

The Purchasing and Stores Department is responsible for the following:

- Purchasing "N" items only from suppliers appearing on the Qualified Suppliers List (QSL).
- Coordinate procurement activities with AEPSC Nuclear Operations and Engineering Divisions, the AEPSC Quality Assurance Department and Cook Plant personnel.
- Prepare and issue requests for quotations, contracts, service orders and purchase orders for "N" items.
- Establish a system to implement corrective action as described in the AEPSC General Procedures for the Cook Plant.
- Establish a system of document keeping, and transmittal.
- Establish a system of document control for controlled procedures, instructions, and purchasing documents for "N" items.
- 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.
- Notify suppliers of their status regarding the QSL, e.g., inclusion, exclusion, conditional approval, etc.
- Notify the Indiana & Michigan Electric company Purchasing Department and the Cook Plant Stores of changes in the QSL.

- Receipt inspection, handling, storage and control of stores items.

1.7.1.2.6 Organization (Cook Plant)

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

- Assistant Plant Manager - Maintenance
- Assistant Plant Manager - Operations
- Administrative Superintendent
- Quality Control Superintendent (reports functionally to the Plant Manager)

The Cook Plant organization, under the Plant Manager 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 Plant Nuclear Safety Review Committee.
- Maintain close working relationships with the NRC as well as local, state, and federal government regulating officials regarding conditions which could affect, or are affected by Cook Plant activities.
- Set up plant load schedules and arrange for equipment outages.
- Develop and efficiently implement all site centralized training activities.
- Direct all facility personnel and safety programs.

- Administer the centralized facility training complex, simulator, and programs ensuring that program development is consistent with the systematic approach to training, INPO, regulatory and corporate requirements.
- Ensure that human resource activities include employee support programs 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, and company policy.
- Supervise, plan, and direct the activities related to the maintenance and installation of all power plant equipment, structures, grounds, and yards.
- Prepare plant maintenance budgets, construction budgets, improvement requisitions, and work orders.
- Prepare and maintain records and reports pertinent to equipment maintenance, cost histories, regulatory agency requirements.
- Administer contracts and schedule outside contractors' work forces.
- Enforce and coordinate plant regulations, procedures, policies, and objectives to assure safety, efficiency, and continuity in the operation of the Cook Plant within the limits of the operating license and the Technical Specifications and formulation of related policies and procedures.
- Plan, schedule, and direct the activities relating to the operation of the Cook Plant and associated switchyards; cooperate in planning and scheduling of work and procedures for refueling and maintenance of the Cook Plant; direct and coordinate fuel loading operations.
- Review reports and records and 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 plant safely and efficiently.

- Approve operator training programs administered by the Cook 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 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 surveillances, and design change installations.
- Coordinate all 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 Program. This encompasses, but is not limited to, the planning and directing of quality control activities to assure that industry codes, Nuclear Regulatory regulations, and company instructions and policies regarding quality control for the nuclear generating station are enforced, and that these activities are properly documented.
- Prepare reports of reportable occurrences which are mandated by the NRC and the Technical Specifications.
- Direct the activities of contractor QC/NDE personnel assigned to the QC Department and provide inspections of work performed.
- Prepare statistical reports utilized in Nuclear Regulatory Appraisal Meetings and Enforcement Conference.
- Coordinate the efforts of outside agencies such as American Nuclear Insurers (ANI), Institute of Nuclear Power Operations (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.
- Qualification and certification of inspection, test, and examination personnel ensuring compliance to Regulatory Guide 1.58, ANSI N45.2.6, the ASME B&PV Code, and SNT-TC-1A, as applicable, except as noted in Appendix B hereto, item 9.
- Conduct of the Quality Control Program, including recommendations for improvement.
- Procurement, receiving, quality control receipt inspection, storage, handling, issue, stock level maintenance, sale, and overall control of stores nuclear and standard grade material, components, and equipment.
- Provide material service and support in accordance with policies and procedures required by AEP Purchasing and Stores, AEPSC Quality Assurance, and the Nuclear Regulatory Commission (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 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 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 D.C. Cook 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 Physical Sciences Sections of the 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 plant's radioactive waste programs, including volume reduction, packaging and shipping.
- Administration of the QA Records Program.

1.7.2 QUALITY ASSURANCE PROGRAM

1.7.2.1 SCOPE

Policies that define and establish the D.C. Cook Nuclear Plant Quality Assurance 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 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, or components 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 structures, systems or components 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 FSAR; or lead to a release of radioactivity in excess of 10CFR100 limits.

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

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

Quality Assurance 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 Quality Assurance Program may be accomplished by AEPSC and/or Indiana & Michigan Electric Company or delegated in whole or in part to other AEP System companies or outside parties. However, AEPSC and/or Indiana & Michigan Electric Company retain full responsibility for all safety-related activities. 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 of AEPSC, as Chief Executive Officer, 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 Donald C. 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. It 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 Quality Assurance Program at AEPSC and the 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 by 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 D.C. Cook Nuclear Plant.

The QA program applies to activities affecting the quality of safety-related structures, systems, components, and related consumables during plant operation, maintenance, testing, and all modifications. Safety-related structures, systems and components are identified in Nuclear (N) Lists and other documents which are developed and maintained for the plant.

1.7.2.2.3

This QAPD, organized to present the Quality Assurance Program for the D.C. Cook Nuclear Plant in the order of the 18 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 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 for compliance with quality-related criteria, and assign responsibility to the various departments, as required, for implementation. Department Head Instructions (DHIs) have been prepared, when required, to implement those activities for each department. Department Head Procedures (DHPs) have been prepared to describe the detailed activities required to support safe and effective plant operation.

The PMIs are reviewed by the AEPSC Supervisor - Quality Assurance (Site) for concurrence that they will satisfactorily implement regulatory requirements and commitments. They are then reviewed by the Plant Nuclear Safety Review Committee (PNSRC) prior to approval by the Plant Manager.

Safety-related DHIs and DHPs are reviewed by the department head of origination, AEPSC Supervisor - Quality Assurance (Site), PNSRC and Plant Manager prior to use.

AEPSC General Procedures (GPs) are utilized to define corporate policies and requirements for quality assurance, and to implement applicable quality assurance requirements within AEPSC.

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

When contractors perform work on-site under their own quality assurance programs, the programs are reviewed for compliance and consistency with the applicable requirements of the Plant's Quality Assurance Program and the contract, and are approved by the AEPSC Supervisor - Quality Assurance (Site), PNSRC and Plant Manager prior to the start of work.

1.7.2.2.5

Provisions of the Quality Assurance Program for the D.C. Cook Nuclear Plant apply to activities affecting the quality of safety-related structures, systems, and components. Appendix A to this QAPD lists the Regulatory Guides and ANSI Standards that identify AEPSC's commitment. Imposition of these guides/standards on AEPSC/I&MECo suppliers and subtier suppliers will be on a case-by-case basis depending upon the item or service to be supplied. 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 D.C. Cook Plant Final Safety Analysis Report (FSAR) for identifying structures, systems and components to which the Quality Assurance Program applies.

b) This identification process results in the N-List for the D.C. Cook Nuclear Plant. This N-List is a controlled document, issued to designated personnel. N-List items are determined by engineering analysis of the function(s) of plant structures, systems and components in relation to safe operation and shutdown.

c) The extent to which controls specified in the Quality Assurance Program are applied to N-List items is determined for each item considering its relative importance to safety. Such determinations are based on data in such documents as the plant Technical Specifications and the FSAR.

1.7.2.2.6

Activities affecting safety 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) Materials and items are in an acceptable status.
e) Systems or structures on which work is to be performed are in the proper condition for the task.
f) Proper 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 are specifically designated, as follows:

a) The Training and Indoctrination Program provides for on-going training and periodic refamiliarization with the Quality Assurance Program for the D.C. Cook Nuclear Plant.

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

c) Personnel who participate in Quality Assurance Audits are qualified in accordance with Regulatory Guide 1.146.

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

e) The Training/Qualification Program includes, as applicable, provisions for retraining, reexamination and recertification to ensure that proficiency is maintained.

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

g) Personnel responsible for performing activities that affect quality are instructed as to the purpose, scope and implementation of the applicable quality related 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 plant personnel is performed employing two techniques, as applicable: 1) on the job and formal training administered by the department or section the individual works for; and 2) formal training conducted by NRC licensed instructors from the Training Department or other entities (internal and external to the AEP System). 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).

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 Quality Assurance Program through Plant Manager and Department Head Instructions and Procedures. These instructions also include a description of the Quality Assurance Program, the use of instructions and procedures, personnel requirements for procedure compliance and the systems and components controlled by the Quality Assurance Program.

1.7.2.2.8

The AEPSC Information System Department (not charted) has established a Computer Software Quality Assurance Section. Procedures are being developed to establish QA requirements for safety-related computer software. The Computer Software QA Section will be subject to periodic audit by the AEPSC QA Department.

1.7.3 DESIGN CONTROL

1.7.3.1 SCOPE

Modifications to structures, systems and components are accomplished in accordance with approved design. Activities to develop such designs are controlled. Depending on the type of modification, these activities include design and field engineering; the performance of physics, seismic, stress, thermal, hydraulic, radiation and Safety Analysis Report (SAR); accident analyses; the development and control of associated computer programs; studies of material compatibility; accessibility for inservice inspection and maintenance; and determination of quality standards. 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

Modifications to the plant are controlled by instructions and procedures. All modifications are reviewed as required by 10CFR50.59.

1.7.3.2.2

A Change Control Board has been established within AEPSC to perform the review and authorization for safety-related design changes [Request for Change (RFCs)]. The Change Control Board is made up of members of the Engineering and Design Divisions within AEPSC.

1.7.3.2.3

Plant originated RFCs are reviewed by the Plant Nuclear Safety Review Committee (PNSRC) and approved by the Plant Manager prior to submission to the Change Control Board. 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. The AEPSC Nuclear Safety and Licensing Section reviews RFCs to determine their impact on nuclear safety and to determine if the proposed changes involve an unreviewed safety question as defined by 10CFR50.59. RFCs are then returned to the PNSRC for subsequent review prior to submission to the Change Control Board. If an RFC were to involve an unreviewed safety question, it would not be approved by the Nuclear Safety and Licensing Section until the required approval was received from the NRC.

1.7.3.2.4

Proposed design changes which require emergency processing are originated at the plant, reviewed by the PNSRC and approved by the Plant Manager. Plant management then contacts the AEPSC Nuclear Operations Division, 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 package.

1.7.3.2.5

When RFCs 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, 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 reviewed 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. Modifications 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.

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.

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

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

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 operational.

1.7.3.2.10

Plant implementation of the RFC is accomplished by the Plant Manager assigning a specific plant department the responsibility for coordinating the design change. 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 cognizant engineer. Further, completed RFCs are reviewed by AEPSC QA (Site) 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

This mechanism provides for: 1) controlled submission of design changes, 2) engineering evaluation, 3) review for impact on nuclear safety, 4) review 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 Quality Assurance Program requirements to the extent necessary to assure adequate quality.

1.7.4.2 IMPLEMENTATION

1.7.4.2.1

Procurement documents for safety-related materials/services originating at the plant, except as denoted below, are processed through AEPSC for review and approval. The plant may request the assistance of AEPSC cognizant engineers in any procurement activity.

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, 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.

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

Each involved manager is responsible for procurement planning, bid solicitation, bid evaluation, and for assuring that the applicable QA requirements are set forth in the procurement documents.

1.7.4.2.2

The N-List, in conjunction with other sources, is used to determine equipment classification. Donald C. Cook Nuclear Plant Specifications (DCC Specifications) are used to determine material and documentation requirements, codes or standards that materials must fulfill, and define the documentation that must accompany the material to the plant.

Department heads cognizant of the equipment and its quality assurance requirements review all procurement documents to assure that correct classification is made; that the appropriate plant specifications which identify quality requirements, are referenced or attached; and that the documentation requirements are properly stated. Purchase requisitions for new safety-related equipment are initiated by the AEPSC cognizant engineers who establish the initial equipment quality assurance requirements. Replacement or spare equipment is procured via the original purchase requirements. In instances where these requirements have been superseded by a revised specification, the replacement/spare part is procured to the revised requirements.

1.7.4.2.3

The contents of procurement documents vary according to the item(s) being purchased and its function(s) in the 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.

c) Regulatory, administrative and reporting requirements.

d) Quality requirements appropriate to the complexity and scope of the work, including necessary tests and 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 Department performs off-line reviews of procurement documents to assure that the procurement documents have been prepared, reviewed and approved per the QA program requirements.

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 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 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.

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

1.7.5.2.2

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

1) General Procedures which are applicable to all divisions of the corporation.

2) Division/Section Procedures which apply to the specific division or section involved.

1.7.5.2.3

The Plant Manager Instructions have been classified into the following series:

1000 Organization
2000 Administration
3000 Procurement, Receiving, Shipping and Storage
4000 Operations, Fuel Handling, Surveillance Testing
5000 Maintenance, Repair and Modification
6000 Technical Services - Chemistry, Radiological Controls, Engineering and Instrument Maintenance and Calibration
7000 Quality Services - Review and Audit, Equipment Classification, Indoctrination and Training, Inspections, etc.

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

Plant drawings are produced, controlled and distributed under the control of AEPSC and the plant. AEPSC design drawings are produced by the AEPSC Design Division under a set of procedures which direct their development and review. These procedures specify requirements for inclusion of quantitative and qualitative acceptance criteria. Specific drawings are reviewed and approved by the cognizant Engineering Divisions.

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 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 designed 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 Section 2.0, "Quality Assurance Program" 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) Final Safety Analysis Report (FSAR).
f) Nuclear Regulatory Commission submittals.
g) Plant Technical Specifications.
h) 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.

Maintenance, modification and inspection procedures are reviewed by AEPSC QA for compliance with established inspection requirements.

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 or for interested parties are appropriately marked to indicate that they are for information use only, and cannot be used to operate or maintain the facility, or to conduct quality-related activities.

1.7.6.2.7

A mechanism has been established which controls responses to NRC documents (I.E. Bulletins, I.E. Inspection Reports, Generic Letters, etc.). These responses, which are uniquely identified by an individual number, require several levels of review and approval.

1.7.7 CONTROL OF PURCHASED MATERIAL, EQUIPMENT, AND SERVICES

1.7.7.1 SCOPE

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

1.7.7.2 IMPLEMENTATION

1.7.7.2.1

AEPSC qualifies suppliers by performing a documented evaluation of their capability to provide items or services specified by procurement documents. All material, equipment and services, designated safety-related, are purchased from suppliers whose QA programs have been accepted in accordance with AEPSC requirements. Qualification of such suppliers and maintenance of a Qualified Supplier List (QSL) is accomplished by the AEPSC QA Department. In the discharge of this responsibility, the AEPSC QA Department utilizes information generated by others (such as the CASE Association and ASME) to aid in the supplier qualification process. Distinction is made between suppliers, stocking distributors (warehouses) and sales offices. The supplier or distributor must be on the QSL before procurement can be completed.

AEPSC is a member of CASE and performs audits for submittal to the CASE Register as well as the plant's Qualified Supplier List. The CASE Register provides a prescreened list of potential suppliers with QA programs. An evaluation is made if there is an interest in a CASE listed supplier to consider the scope of the qualification audit and the identity of the auditor which are stated in the Register. Additional program surveys will be conducted, as necessary, to meet requirements.

Acceptance is not complete until it has been determined that the supplier can meet the basic QA and technical requirements of the product or service that is required.

1.7.7.2.2

For commercial "off-the-shelf" items where the requirements for a specific Quality Assurance Program appropriate for nuclear applications cannot be imposed in a practical manner, source verification is used to provide adequate assurance of acceptability.

1.7.7.2.3

In-process surveillance of suppliers' activities during fabrication, inspection, testing and shipment of items is performed when deemed necessary, depending upon supplier qualification status, complexity and importance to safety of the item being furnished, and/or previous supplier history. This surveillance is performed by the cognizant engineering department, responsible plant department, or AEPSC QA, or any combination thereof.

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, interchangeability, safety, 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 materials addresses the program for inspection of incoming materials 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. Receipt inspection provisions apply regardless of whether procurement originates at the plant or at AEPSC. Additional inspections may apply if required by the procurement document.

Where materials and/or services are safety-related and procurement is accomplished without assistance of AEPSC, supplier selection is limited to those companies identified on the Qualified Suppliers List (QSL).

1.7.7.2.6

Materials received at the site are tagged with a "Hold" tag and placed in a designated, controlled area 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 material is placed in a designated area of the storeroom. Material traceability to procurement documents and to end use is maintained through recording of Hold Tag and Acceptance Tag number on applicable documents.

Nonconforming materials, or missing or questionable documentation results in materials being kept on hold and placed in a designated, controlled area of the storeroom. If the nonconformance cannot be cleared, the material 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 plant quality assurance program and procedures. Prior to their working at the plant, contractor quality assurance programs and procedures must be reviewed and approved by the AEPSC Site Quality Assurance Supervisor, PNSRC and the Plant Manager. Further, periodic audits of site contractor activities are conducted under the direction of the AEPSC Site Quality Assurance Supervisor.

1.7.7.2.8

Suppliers are required to furnish the following records:

a) Applicable drawings and related engineering documentation that identify the purchased item and the specific procurement requirements (e.g., codes, standards and specifications) met by the item.

b) Documentation identifying any procurement requirements that have not been met.

c) A description of those nonconformances from the procurement requirements dispositioned "use-as-is" or "repair".

d) Quality records as specified in the procurement requirements.

1.7.7.2.9

The validity of supplier certificates of conformance is evaluated at the time of supplier resurvey and requalification, and is based on the continual implementation of the supplier's QA program.

1.7.8 IDENTIFICATION AND CONTROL OF ITEMS

1.7.8.1 SCOPE

Materials, parts and components (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 materials, parts and components (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, or serial number are included in the specifications and/or purchase order.

1.7.8.2.5

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

1.7.9.2.2

Special process requirements for chemical cleaning, application of protective coatings and concrete placement are set forth in AEPSC Specifications 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 reviewed by the AEPSC QA Department.

Special process requirements for welding, heat treating and nondestructive examination (NDE) are set forth in AEPSC Specifications and the AEPSC Welding and NDE Manuals. These specifications and manuals are prepared by or are reviewed and approved by the AEPSC Staff Engineer - Chief Metallurgist (Corporate NDE Level III Administrator). The AEPSC NDE Manual is reviewed by the AEPSC QA Department.

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

Welding is performed in accordance with the procedure contained in the AEPSC Welding Manual. These procedures are qualified in accordance with applicable codes and standards, and Procedure Qualification Records are prepared. The weld procedure qualification record is reviewed and approved by the AEPSC Staff Engineer - Chief Metallurgist. Weld qualification documentation is retained in the AEPSC Welding Manual.

Contractor welding procedures are qualified by the contractor. These procedures and the qualification documentation are reviewed and approved by the plant and the AEPSC Staff Engineer - Chief Metallurgist. This documentation is retained by the contractor.

1.7.9.2.3

Nondestructive examination personnel are qualified and certified by either a designated Corporate NDE Level III Administrator or by a Cook Plant NDE Level III (NDE Supervisor) who has been qualified and certified by the designated Corporate NDE Level III Administrator. Certification is by examination. Personnel qualification is kept current by performance of the special process(es) and/or reexamination at time intervals specified by the AEPSC NDE Manual. Unsatisfactory performance or, where applicable, failure to perform within the designated time intervals, requires recertification.

Plant welders are qualified by the Maintenance Department utilizing the procedures in the AEPSC Welding Manual. Examination of specimens is performed by the QC Department in accordance with the AEPSC Specification covering welder qualification. Plant welder qualification records are maintained for each welder by the Maintenance Department. Contractor and craft welders are qualified by the contractor utilizing procedures approved by the plant and the AEPSC Staff Engineer - Chief Metallurgist. Contractor and craft welder qualification records are maintained by the contractor.

1.7.9.2.4

Quality Control Technicians assigned to the Quality Control Department perform nondestructive testing for work performed by plant and contractor personnel. These individuals are qualified to SNT-TC-1A and records of the qualifications are maintained at the 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 Section 1.7.17, "Quality Assurance Records".

1.7.9.2.7

The documentation resulting from welding and nondestructive testing is reviewed by appropriate management 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 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, and the individual(s) by job title who must perform or attest to 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 and Qualification Programs for personnel who perform inspections are established, implemented and documented in accordance with Section 1.7.2, "Quality Assurance Program".

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.

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

The Plant Quality Control Department has been assigned the responsibility for establishing and executing the following programs:

a) In-process verifications and inspections.
b) Inservice inspections.

To ensure the quality of the maintenance, operation, technical, administrative, planning and construction activities at the D.C. Cook Nuclear Plant, the Plant Quality Control Department will inspect, monitor and verify key attributes that have been deemed necessary to assure the acceptability of:

a) Equipment
b) Tests
c) Processes
d) Materials
e) Parts
f) Components
g) System checks

The performance of these inspections, verifications and monitoring will be defined by instructions/procedures written by the responsible plant departments.

1.7.10.2.5

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.

1.7.10.2.6

Inspection of work associated with normal operation of the plant, such as surveillance tests and verification of routine maintenance, may be performed by individuals in the same group as that which performed the work, but not by personnel who directly performed or supervised the work. The qualification of these personnel is described in Appendix B hereto, item 9b, with exceptions as noted therein.

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. The scheduling of these activities is reviewed by an Assistant Plant Manager. 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 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

Each involved plant department has established procedures for calibration and control of measuring and test equipment utilized in the measurement, inspection and monitoring of structures, systems and components. These procedures describe calibration techniques and frequencies, and maintenance and control of the equipment.

The AEPSC Site Quality Assurance Section 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 utilizing labels which are to be attached to measuring and test equipment to display the date calibrated and the next calibration due date. Where labels cannot be attached, 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 where they exist. Where national 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 that the equipment being calibrated will be within the required tolerances and the basis for acceptance is documented and authorized by the responsible manager.

1.7.12.2.5

A series of PMIs 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. Storage procedures also provide methods to assure that specified shelf lives are not exceeded.

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 plant. Controls are established in PMIs.

1.7.13.2.3

Packaging and shipping requirements are provided to vendors with the DCC Specifications which are a part of the purchase order. Controls for receipt inspection, damaged items and special handling requirements at the plant are established by a PMI. 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 RFC (Design Change) activities, including item fabrication, installation and test, a PMI 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 PMI establishes a tagging system for bypassed safety functions. For those items requiring calibration, a PMI exists which requires physical indication of calibration status by calibration stickers.

1.7.14.2.2

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

The inspection status of materials received at the plant is identified in accordance with instructions established in a PMI. 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 supervisory personnel inspect an activity at various stages and sign off the procedural steps covered by the inspection.

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

1.7.14.2.3

Required surveillance test procedures are defined in a PMI. This instruction provides for documenting bypassed tests, and for rescheduling of the test. An Assistant Plant Manager reviews the completed and signed off Weekly Surveillance Test Schedule to assure compliance.

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. In situations when an immediate procedure change is required to continue in-process work or testing and the required complete review and approval process can not 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 MATERIALS, PARTS, OR COMPONENTS

1.7.15.2 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 Report System (refer to Section 16.0) are used at D.C. Cook Nuclear Plant to identify nonconforming items and initiate corrective action. Systems, components, or materials which require repair or inspection are controlled under the Job Order System. In addition, the various procedures identified in Section 14 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 Department 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 conditionally 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, and corrective action is taken to preclude repetition. 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 Department.

AEPSC accomplishes corrective action in the following manner:

a) Audit reports which require action as a result of a corrective action request.

b) In accordance with established procedures for Condition Reports, Noncompliance Reports, Inspection Reports and Audit Reports.

c) As required by NRC Letters, I.E. Bulletins and Inspection Reports.

d) As required by 10CFR, Part 21 identified deficiencies.

1.7.16.2.2

Condition Reports provide the mechanism for plant personnel to notify management of conditions adverse to quality. Investigations of reported conditions adverse to quality are assigned by management. The investigation report is used to identify the need for changes to instructions or procedures, the initiation of a design change to correct system or equipment deficiencies, or the initiation of job orders to correct minor deficiencies. Further, Condition Reports are used to identify those actions necessary to prevent recurrence of the reported condition. Condition Reports are also used to report violations to codes, regulations and the Technical Specifications. Condition Reports are reviewed by the PNSRC for evaluation of actions taken to correct the deficiency and prevent recurrence.

Noncompliance Reports (NCRs) provide the mechanism for AEPSC personnel to identify noncompliances. Investigation of reported conditions are assigned to the responsible individual. NCR investigation requires the determination of the cause of the condition and identification of immediate action and action taken to prevent recurrence.

The AEPSC Nuclear Operations Division receives copies of Condition Reports for distribution, on a selected basis, to cognizant engineering departments for review.

The AEPSC Nuclear Safety and Design Review Committee reviews Condition Reports, NCRs, NRC Inspection Report Responses, 10CFR21 items and QA and NSDRC audits for independent evaluation of the reported conditions and corrective actions.

The QA Department 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 quality 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) Nonconformance reports.
h) Corrective action reports.
i) Other documentation such as drawings, specifications, procurement documents, calibration procedures and reports.

1.7.17.2.2

Instructions and procedures establish the requirements for the identification and preparation of records for systems and equipment under the Quality Assurance Program, and provides 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 Accounting 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 establishing procedures for the identification, collection, maintenance and storage of records generated by his department/division. These procedures shall 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.

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, 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 Quality Assurance 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 are reviewed by management. Follow-up action is taken where indicated.

1.7.18.2 IMPLEMENTATION

1.7.18.2.1 AEPSC QA Department Responsibilities

The basic responsibility for the assessment of the Quality Assurance Programs is vested in the AEPSC QA Department. They are primarily responsible for ensuring that proper QA programs are established and implemented. These responsibilities are discharged in cooperation with the AEPSC and plant management, and their staffs.

Stop Work Authority - Refer to Section 1.7.1.2.5 herein.

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 two years.

1.7.18.2.3

The AEPSC Quality Assurance Department conducts audits to verify the adequacy and implementation of the QA Program at the plant and within AEPSC. QA audit reports are distributed to the Plant Manager and PNSRC (site audits) and the NSDRC (all audits).

1.7.18.2.4

The independent off-site review and audit organization is the AEPSC Nuclear Safety and Design Review Committee (NSDRC). This committee is composed of AEPSC, I&M and plant management members. A Charter and

Procedures Manual has been developed for this committee. The NSDRC conducts periodic audits of plant operations pursuant to established criteria (Technical Specifications, etc.).

NSDRC Audit Reports are submitted for review to the Chairman of the NSDRC and to the Vice Chairman Engineering and Construction. Corrective Action Requests provide for the recording of actions taken to correct deficiencies found during these audits.

1.7.18.2.5

The plant on-site review group is the Plant Nuclear Safety Review Committee (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 and procedures for safety-related systems prior to approval by the Plant Manager. In addition, this committee serves to conduct investigations of violations to Technical Specifications, reviews Condition Reports to determine if appropriate action has been taken and reviews all design changes.

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 foregoing scheduling criteria.

1.7.18.2.7

Regularly scheduled audits are supplemented by special audits when significant changes are made in the Quality Assurance 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 quality related practices, procedures, instructions, activities and items; and 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 Quality Assurance 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 Quality Assurance Organization, but knowledgeable in auditing and quality assurance, periodically review the effectiveness of the Quality Assurance 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 D.C. Cook Nuclear Plant Fire Protection QA Program has been developed using the guidance of the NRC Branch Technical Position 9.5-1, Appendix "A".

This QA Program is applicable to:

1) Fire protection areas and equipment designed and/or procured after January 31, 1977 that protects safety-related items which appear in the Fire Protection Technical Specifications; and,

2) The balance of plant fire protection areas and equipment designed and/or procured after January 31, 1977.

Implementation of the Fire Protection QA Program is the responsibility of each involved AEP organization.

The QA Program for the Fire Protection Program at D.C. Cook Plant applies to the following activities: design, procurement, fabrication, construction, operation, maintenance and modification.

1.7.19.2 Organization

The QA program for fire protection is under the management control of AEPSC. This control consists of:

1) Formulating and verifying that the Fire Protection QA Program incorporates suitable requirements and is acceptable to the management responsible for fire protection; and,

2) Verifying the effectiveness of the QA program for fire protection through review, surveillance and audits. The QA program for fire protection is part of the overall plant QA program. These QA criteria apply to those items within the scope of the Fire Protection Program, such as fire protection systems, emergency lighting, communication and emergency breathing apparatus, as well as the fire protection requirements of applicable safety-related equipment.

AEPSC and plant management has direct functional responsibility for the formulation, implementation and assessment of the D.C. Cook Fire Protection Program.

The Section Manager - Fire Protection and HVAC and the Fire Protection Engineer have coordinated the building layout, the fire suppression and fire detection systems, commensurate with fire areas within the plant. They have established the design of the overall fire detection/suppression system and the incremental parts of the system. Maintenance information has been provided to the plant in the form of system descriptions and equipment supplier instruction material.

The Fire Protection Program at the D.C. Cook Plant provides for inspection of fire and explosion hazards and training of fire brigades and responding fire departments. The Plant Manager has delegated responsibility to various plant departments for the following fire protection activities:

a) Maintenance of fire protection system,
b) Testing of fire protection equipment,
c) Fire safety inspections,
d) Fire fighting procedures, and
e) Fire drills.

The Shift Supervisor on duty is designated as the Fire Chief and coordinates the fire fighting efforts of shift personnel and the fire brigade.

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 deviations and changes from these quality standards are controlled.

The plant design was reviewed by qualified personnel to assure inclusion of appropriate fire protection requirements. These reviews include items such as:

1) Reviews to verify adequacy of wiring isolation and cable separation criteria.

2) Reviews to verify appropriate requirements for room isolation (sealing penetrations, floors and other fire barriers).

3) Reviews to determine increase in fire loadings.

4) Reviews to determine the need for additional fire detection and suppression equipment.

A review and concurrence of the adequacy of fire protection requirements and quality requirements stated in procurement documents is performed. This review determines that fire protection requirements and quality requirements are correctly stated, verifiable and controllable; there are adequate acceptance and rejection criteria; and the procurement document has been prepared, reviewed and approved in accordance with QA program requirements.

Design and procurement document changes, including field changes and design deviations are subject to the same level of controls, reviews and approvals that were applicable to the original document.

1.7.19.4 Instructions, Procedures and Drawings

Inspections, tests, administrative controls, fire drills and training that govern the Fire Protection Program are prescribed by documented instructions, procedures, or drawings, and are accomplished in accordance with these documents.

Indoctrination and training programs for fire prevention and fire fighting are implemented in accordance with documented procedures. Activities associated with the fire protection system are prescribed and accomplished in accordance with documented instructions, procedures and drawings.

Instructions and procedures for design installation, inspection, test, maintenance, modification and administrative controls are reviewed to assure that proper fire protection requirements are included.

1.7.19.5 Control of Purchased Material, Equipment and Services

Measures are established to assure that purchased material, equipment and services conform to the 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 receiving inspections.

Source or receiving inspection is provided, as a minimum, for those items whose 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: 1) installation, maintenance and modification of fire protection systems; and 2) emergency lighting and communication equipment.

b) Inspections of penetration seals and fire retardant coating installations to verify the activity is satisfactorily completed.

c) Inspections of cable routing to verify conformance with design requirements.

d) Inspections to verify that appropriate requirements for room isolation are accomplished following construction or modification 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 and/or check lists are provided for inspections.

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

h) Periodic inspections of materials subject to degradation such as fire stops, seals and fire retardant coating.

1.7.19.7 Test and Test Control

a) Installation testing - Following installation, modification, repair, or replacement, sufficient testing is performed to demonstrate that the fire protection systems, emergency lighting and communication 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 schedules and methods have been implemented and the results documented. Fire protection equipment, emergency lighting and communication equipment are tested periodically to assure that the equipment functions properly.

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 cualified responsible individual or group.

1.7.19.8 Inspection, Test and Operating Status

The inspection, test and operating status for the Fire Protection System are performed as described in Section 1.7.14.

1.7.19.9 Nonconforming Items

Nonconforming items for the fire protection components are identified and dispositioned as described in Section 1.7.15.

1.7.19.10 Corrective Action

The corrective action mechanism described in Section 1.7.16 applies to the fire protection system.

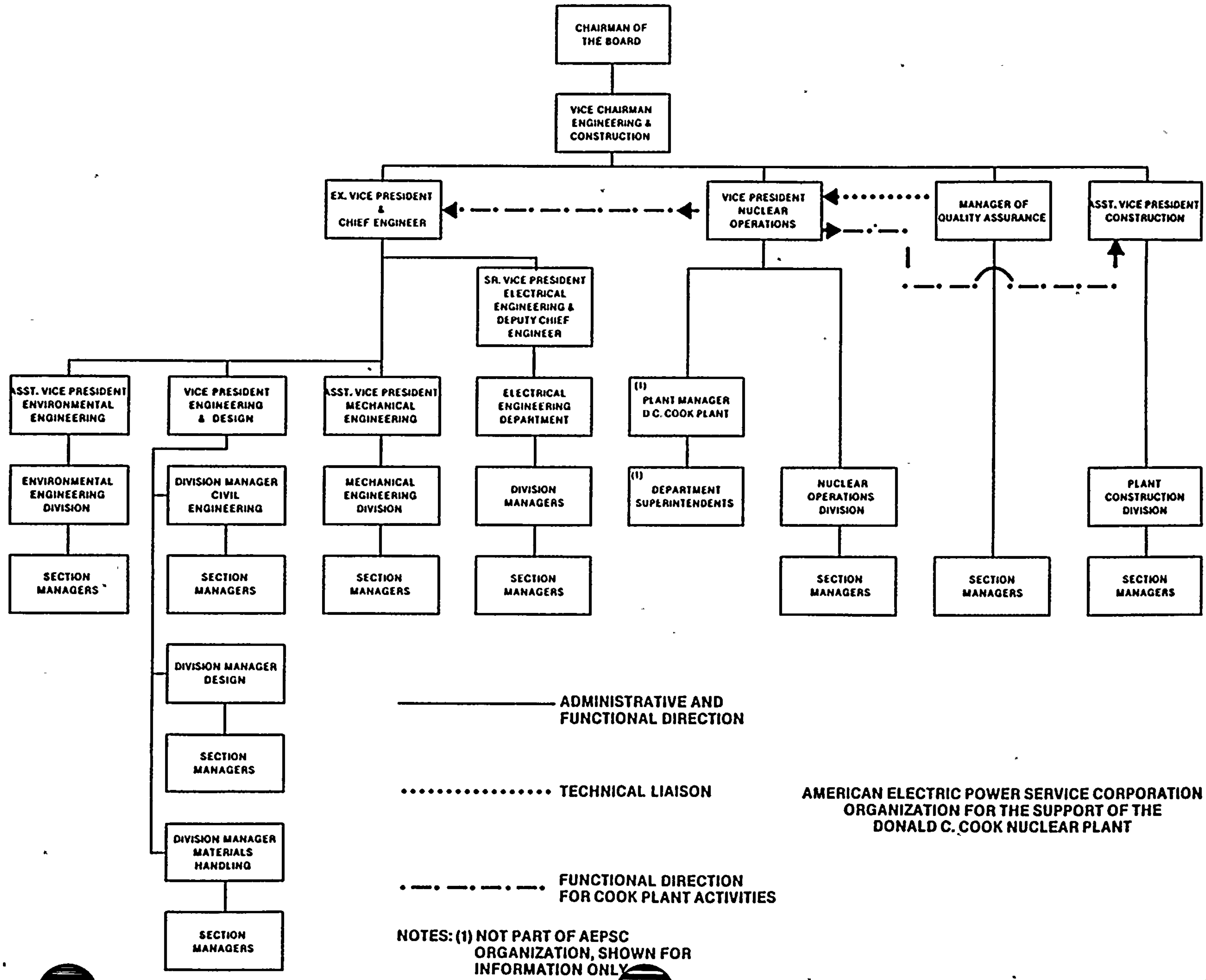
1.7.19.11 Records

Records generated to support the fire protection system and its components are controlled as described in Section 1.7.17.

1.7.19.12 Audits

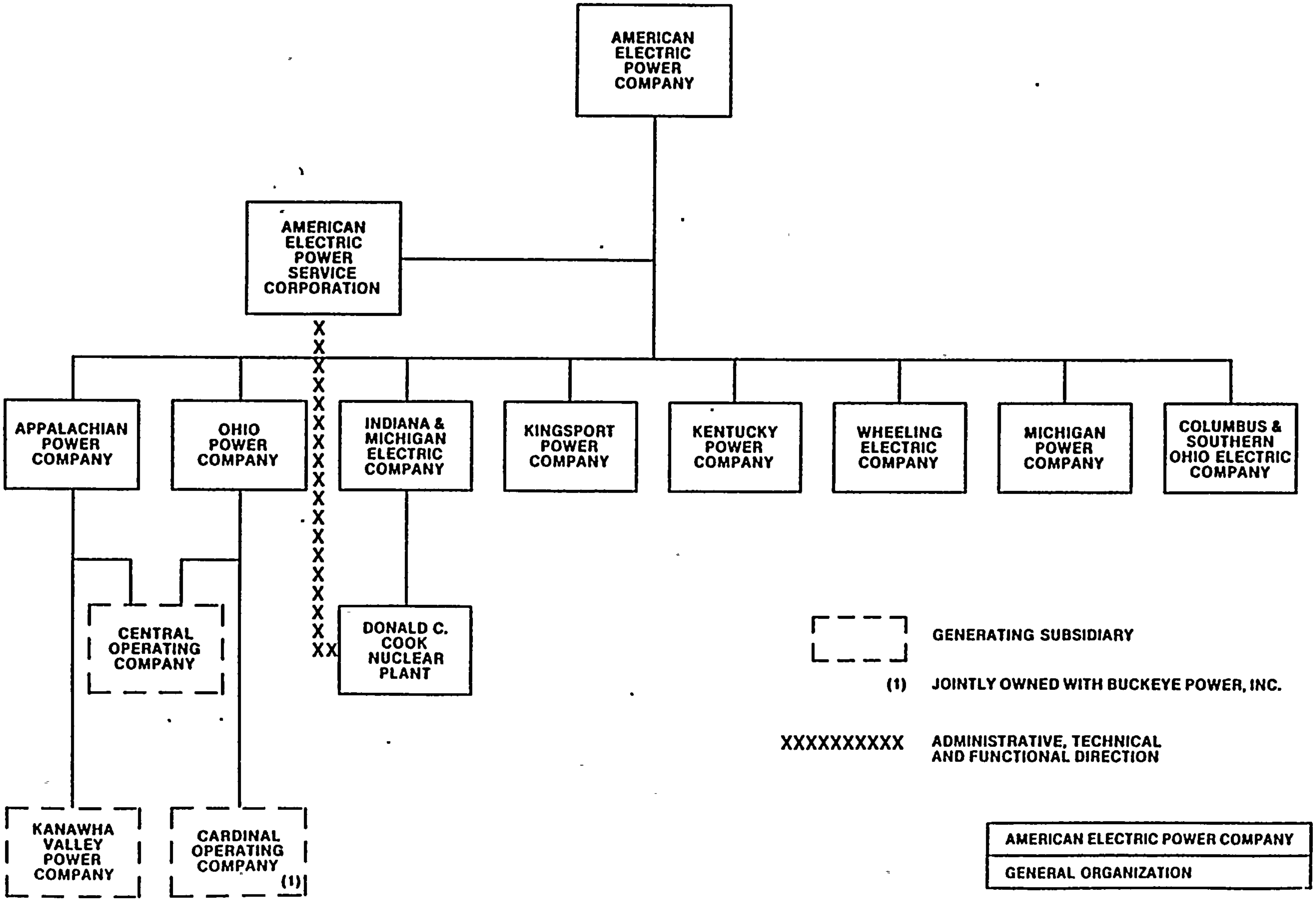
Audits are conducted and documented to verify compliance with the Fire Protection Program as described in Section 1.7.18.

Audits are periodically performed to verify compliance with the administrative controls and implementation of quality assurance criteria. The audits are performed in accordance with preestablished written procedures or check lists. 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
ORGANIZATION FOR THE SUPPORT OF THE
DONALD C. COOK NUCLEAR PLANT

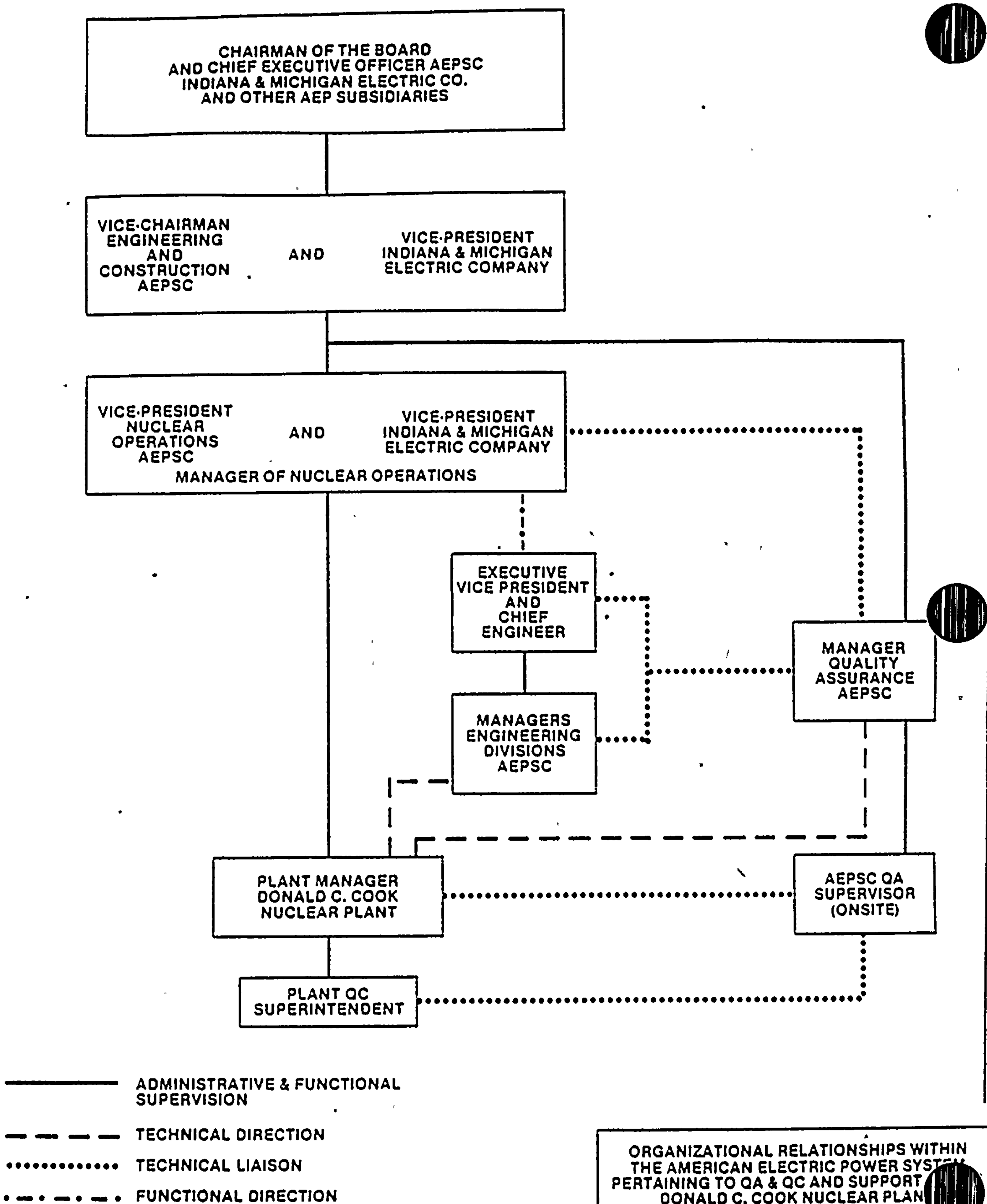
Figure 1.7-1



AMERICAN ELECTRIC POWER COMPANY

GENERAL ORGANIZATION

Figure 1.7-2



ORGANIZATIONAL RELATIONSHIPS WITHIN THE AMERICAN ELECTRIC POWER SYSTEM PERTAINING TO QA & QC AND SUPPORT DONALD C. COOK NUCLEAR PLANT

QA ORGANIZATION

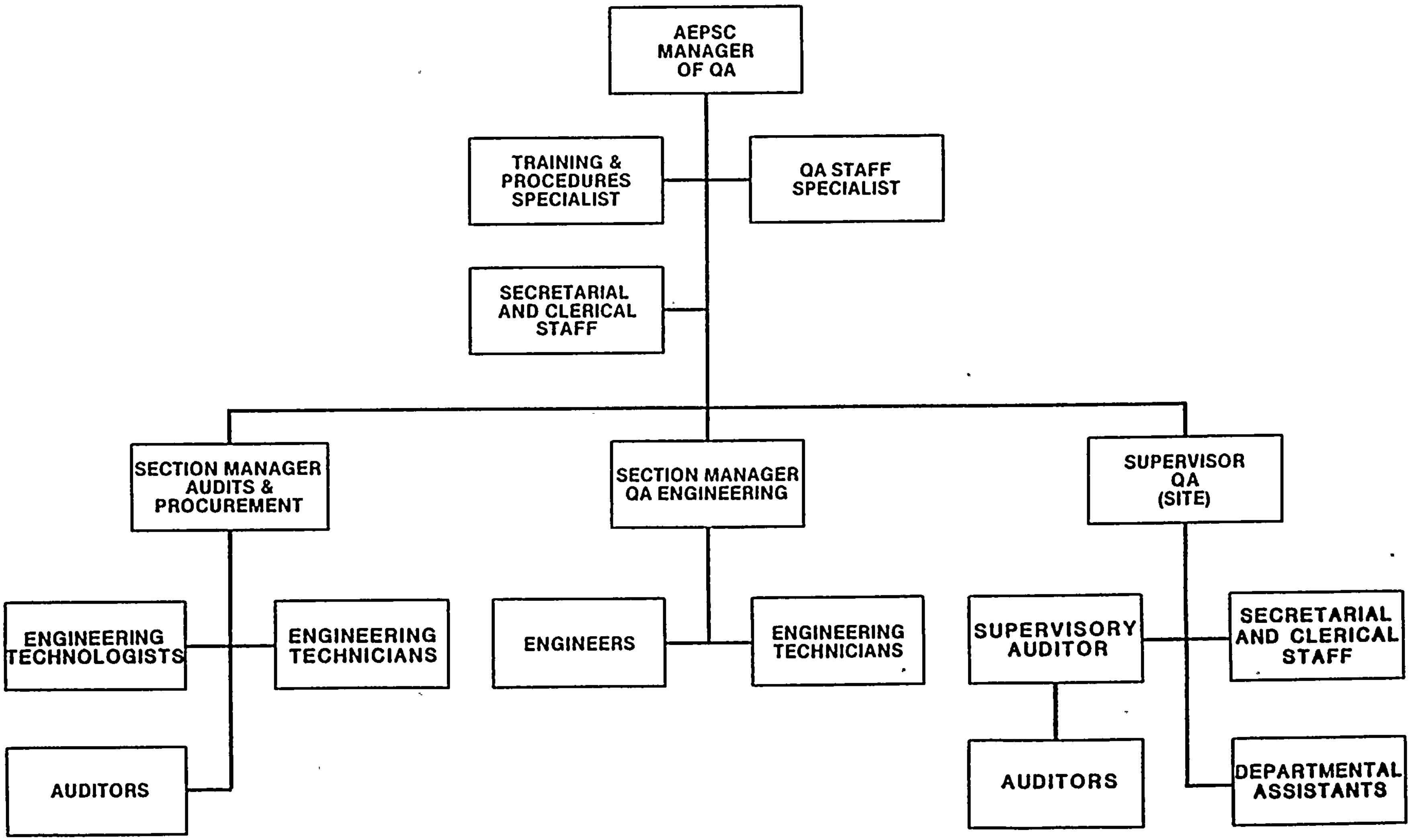
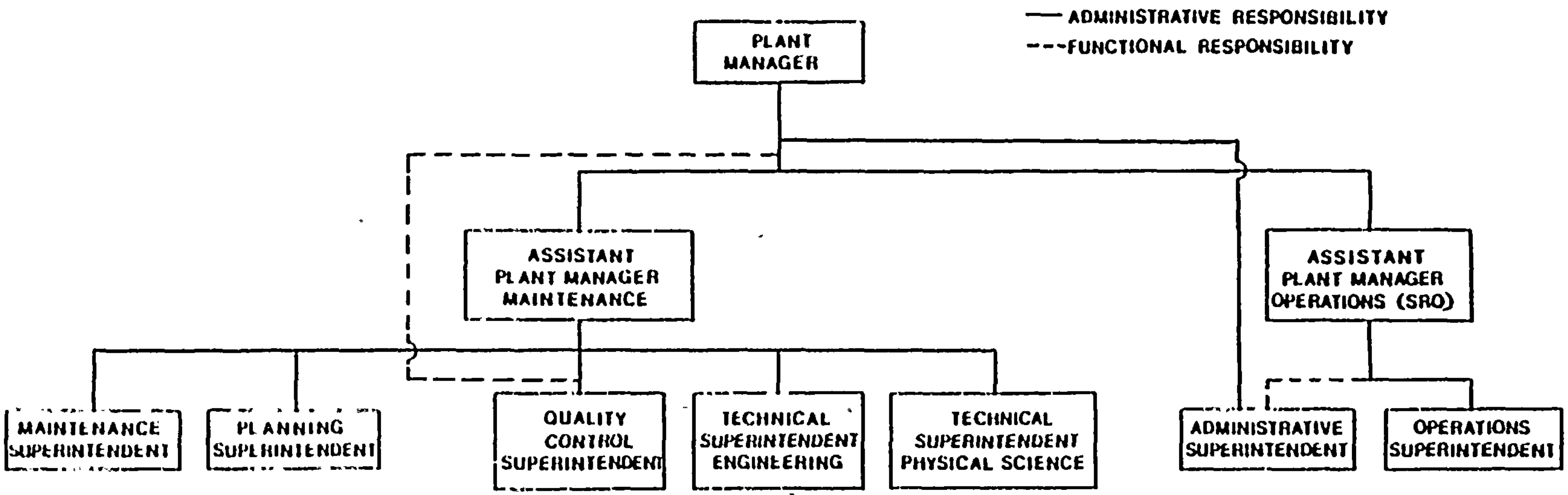


Figure 1.7-4



INDIANA & MICHIGAN ELECTRIC COMPANY
ORGANIZATION
FOR THE
DONALD C. COOK NUCLEAR PLANT

Figure 1.7-5

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.	Safety Guide 33, Appendix A (11/72)	-	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 1.38 (10/76)	-	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants
	ANSI N 45.2.2 (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 Strucutral 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)	-	Requiremens 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
22.	ANSI N510 (1975)	-	Testing of Nuclear Air-Cleaning Systems

AEPSC/I&MECO 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&MECo 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&MECo 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 alternatives 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

Requirement

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

Exception/Interpretation

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

Requirement

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

Exception/Interpretation

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

2e. Sec. 4.3.4(3)

Requirement

"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."

Exception/Interpretation

The NSDRC and PNSRC will not review Technical Specification changes after NRC approval prior to implementation. The basis for this position is the NSDRC and PNSRC review Technical Specification changes prior to submittal to the NRC.

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&MECo 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&MECo 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. 5.2.8

Requirement

"A surveillance testing and inspection program . . . shall include the establishment of a master surveillance schedule reflecting the status of all planned in-plant surveillances 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 - 1971."

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. 5.2.16

Requirement

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

Exception/Interpretation

See Item 6b.

2m. 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. 2

Requirement

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

Exception/Interpretation

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

Cleanness is maintained, consistent with the work being performed, so as to prevent the introduction of foreign material. As a minimum, cleanness 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&MECo 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

Requirement

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

Exception/Interpretation

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

Requirement

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

Exception/Interpretation

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

Requirement

"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."

Exception/Interpretation

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 that 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

Cleanness 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&MECo 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&MECo 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. Reg. Guide 1.58 - General

Requirement

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

9a. C.2.6

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&MECo 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&MECo 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&MECo does not implement the recommended "Level" designation guidelines.

It is I&MECo'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&MECo maintains that it meets the essence of these training guidelines. The preparation of leak test procedures and the conduct of leak tests at Cook Plant is under the direct supervisor 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&MECo 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 routing, informal orientation on testing programs, to ensure that these individuals fully understand the requirements of performing a leak test.

9b. C5, C6, C7, C8, C10

Exception/Interpretation

I&MECo 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&MECo 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&MECo 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 Donald C. 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 18.1 which exceeds those of ANSI N45.2.6. In I&MECo's opinion, no useful purpose is served by classification of management, supervisory and engineering personnel into "Levels."

Therefore, I&MECo takes the following positions relative to regulatory positions C5, 6, 7, 8 and 10 of Regulatory Guide 1.58.

C-5 Based on the discussion in B.1 above, this position is not applicable to the Donald C. Cook Nuclear Plant.

C-6 Replacement personnel for Donald C. 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&MECo, as a general practice, complies with the training recommendations as set forth in this regulatory position.

C-8 All I&MECo 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&MECo 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. <u>N45.2.8,</u>

10a. Sec. 2.9e

<u>Requirement</u>

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.

<u>Exception/Interpretation</u>

All of the documents listed are not necessarily required <u>at</u> the construction site for installation and testing. AEPSC and I&MECo assure that they are available <u>to</u> the site as necessary.

10b. Sec. 2.9e

<u>Requirement</u>

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

<u>Exception/Interpretation</u>

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

<u>Requirement</u>

"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.)

<u>Exception/Interpretation</u>

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 Donald C. Cook Nuclear Plant Master File Room and other off-site record storage facilities comply with the requirements of NUREG-0800 (7/81), Section 17.1.17.4.

12. Reg. Guide 1.144,

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&MECo considers that the National Bureau of Standards or other State and Federal Agencies which may provide services to AEPSC/I&MECo are not required to be audited.

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&MECo 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. 3.3.2

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&MECo establishes post-installation test requirements giving due consideration to supplier recommendations.

14. Reg. Guide 1.58/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 Manager of QA who is a qualified lead auditor; and is administered by designated QA Department Section Managers who are also qualified lead auditors.

Audits are, in most cases, conducted by individual auditors, not by "audit teams". These auditors are qualified by established procedures and are assigned by the responsible QA Section Manager 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 Department Section Managers, thereby meeting the requirements of ANSI N45.2.23 relative to "Lead Auditors", and "Audit Team Leaders".

The inland satellite and the beach satellite are also no longer in operation.

New Meteorological Instruments

The current meteorological instrumentation is located east of the plant on the microwave tower. The tower contains the following redundant instrumentation:

180 ft. level	-	Temperature Sensors
150 ft. level	-	Wind Speed and Direction Sensors
50 ft. level	-	Wind Speed and Direction Sensors
30 ft. level	-	Temperature Sensors.

Additionally, a Dew Point and Precipitation Sensor are provided. This instrumentation provides readouts in both control rooms, and the data is recorded in Unit 1 control room.

The base USGS elevation of the microwave tower is 735'-0".

Special Studies

Phenomena having relatively long recurrence intervals, such as tornadoes and ice storms, in the area cannot be studied directly from site observations and estimates have been derived from special reports.(1, 2, 3, 4)

Analysis

The meteorological data from the Donald C. Cook Nuclear Plant site have been abstracted, processed and analyzed on a monthly basis by Smith-Singer, Meteorologists, Inc. The computer output from which the analysis is made is too extensive to include as a part of this report. The summaries given here are derived from it. Table 2.2-1 is a sample of the original hourly records in the computer data file.

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.

Temperatures, Precipitation, Humidity and Barometric Pressure

These elements are largely of value in the general engineering design. The temperature and precipitation data reported in Tables 2.2-2 and 3 have been obtained from the plant site.

High Winds

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 inches of mercury 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. The 1° latitude-longitude square containing Benton Harbor has had 13 tornadoes between 1916 and 1961 while some sectors in states to the southwest have had 70 to 90. This frequency of occurrence can be translated (after Thom)[3] into a probability of a tornado affecting the site once in 1042 years.

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 1898 to 1965, 33 significant ice storms have been reported in this area.

2.2.3 DISPERSION METEOROLOGY

The micrometeorology of the site seems fairly typical of the northern lake regions. The sand dunes in the immediate vicinity cause some aberration of wind flow at low levels for short distances but, in general, the wind is vigorous, turbulent and uncomplicated over the entire area. The thermal stability shows approximately the seasonal variation expected close to large lakes, exhibiting almost no stable cases during the winter months, contrasted with a slightly greater frequency in inversions in the late spring and summer when the air temperature is usually warmer than that of the lake surface. Even in the least favorable month, however, the inversion frequency is only 22%. There are almost no instances in which stable lapse rates are accompanied by winds toward the heavily populated Chicago area.

Turbulence Classifications

It is helpful in studies of dispersion climatology to have a single parameter indicative of the general turbulence to serve as a reference. Neither thermal stability nor wind speed can be used alone as such an indicator, although both are closely related to turbulence, but a qualitative classification can be made directly from the 200 ft. level on the tower.

The four turbulence classes employed in this analysis follow closely the system developed and used extensively by Smith and Singer. The classification is defined in Figure 2.2-3 together with sample wind direction traces. Table 2.2-4 shows the distribution of the four turbulence classes on a monthly and an annual basis. Individual monthly variations among the three years were small, and the overall summary is a good representation of the typical distribution.

It is important to relate the turbulence classes to other parameters representing the atmospheric dispersive capacity of the site. The most direct and significant relationship is developed from the fluctuations of a bi-directional wind vane. Perhaps the most convincing evidence of the strength and turbulence of the wind flow is the fact that the bivane installed on the 200-foot meteorological tower has been continually damaged by the continuous exposure. However, some early bivane data have been collected and analyzed. The bivane was removed from service in early 1969.

The turbulence classes can be compared to the lapse rate and wind speed distributions and this has been done.

From Table 2.2-4 it is seen that the typical daytime turbulence (Class II) dominates the distribution throughout the year, accounting for 81% of all hours and never occurring less than 70% of the time in any month. The surprisingly small frequency of stable (Class IV) conditions which was noted in the PSAR is apparently a genuine feature of the site, since its annual average occurrence is only 7% of the total. The most marked tendency for stable conditions occurs in the summer when the general wind flow in the area becomes relatively light. There is a tendency for an increase in the number of stable hours during the spring, when the lake water is cold compared to the air temperature, but it is not especially marked.

Lapse Rates

The temperature lapse rates between the 200 and 50-foot levels on the main tower are summarized in Table 2.2-5. Most of the months shown represent two years of data rather than three because of malfunctions of the instrumentation at certain times, but the data given are a fair approximation of the thermal stability.

A difference is noted between the turbulence class data of Table 2.2-4 and the stability as represented by the lapse rates in Table 2.2-5. Approximately 20% of the hours of Table 2.2-5 have inversions, whereas only 7% of the turbulence class data appear to be stable. The difference is in part attributable to difficulties with the instrumentation, but it probably also has to do with the possibility of having significant turbulence with slightly positive lapse rates, if the wind speed is strong and the terrain rough. Since the turbulence class measurement is a direct indication of the fluctuation of a wind instrument, we consider it to be a more reliable measure of turbulence.

Turbulence Classes in Association with Lapse Rates and Wind Speeds

Tables 2.2-6 through 10 summarize the relationship among lapse rates, wind speeds and turbulence classes, as well as providing an overall view of the wind speed and lapse rate distributions.

Turbulence Class I represents a very small percentage of the total observational period, as has been noted in Table 2.2-4, and it clearly is related to unstable lapse rates. Turbulence Class II (Table 2.2-7) is also primarily related to unstable lapse rates, but a significant portion of the cases are associated with stable lapse rates and strong wind speeds. The stormy conditions are represented by Class III, with most of the cases in Table 2.2-8 appearing in conjunction with high winds. The Class IV condition has an unusually wide scatter in its relation with both winds and lapse rates. Many, but by no means the majority, of the cases are found with inversions (Table 2.2-9) but light winds and instability also account for a number of Class IV types. This association is almost certainly a result of the flow direction from the lake, which can be lacking in turbulence if the winds are light and the lake surface reasonably calm.

Table 2.2-10 summarized the relation between lapse rates and wind speeds for all hours, regardless of turbulence class. The remarkable feature of the table is the wind distribution. Thirty-four percent of the winds at the 200-foot level exceed 18 mph, and only 3% fall into the 0-3 mph category. The lapse rate distribution, as has been noted earlier, is not especially unusual for a site with such strong winds. One notes some difference in the total percentages within lapse rate groups between Table 2.2-5 and Table 2.2-10. This results from the inclusion of all hours having acceptable lapse rate data in the former, whereas only those hours with both good wind and lapse rate data are included in Table 2.2-10.

<u>Wind Direction and Speed Distributions</u>

There are many systems of summarizing wind data for a study of site dispersion characteristics, but the key questions are usually defined as the most probable annual dose rates from the small quantities of gaseous effluents released in normal operation and estimation of the least favorable conditions that might follow an accident. In the following sections, tabular data are presented that can be translated directly in terms of these two questions, but illustrative material is also included to provide a clearer picture of the site characteristics.

Wind Rose Associated with Turbulence Classes

Figures 2.2-4 through 18 comprise a series of annual wind roses arranged in accordance with turbulence classes for the 200 and 50-foot height on the tower and for the inland satellite location. They are plotted as the percentage of all hours observed and they are arranged in consecutive sets of three according to turbulence class.

Turbulence Class I

Figures 2.2-4, 5 and 6 are the Class I wind roses for the 200-foot, 50-foot and inland satellite locations, respectively. They are noteworthy only in that the percentage of such cases is very small and that the prominent wind directions are S and SE.

Turbulence Class II

The wind roses for this classification deserve somewhat more detailed attention, since this turbulence regime represents 80% of all hours. At all three locations (Figures 2.2-7, 8 and 9) the wind direction distributions are marked in the variety of directions represented. Peaks are noted from the NNW, WNW, and SSW and only the NE direction sector is relatively neglected. The peaks in the distribution are not outstanding, however, generally ranging from 3 to 4% in 10-degree intervals from the most prominent direction sectors. It is also noteworthy that the mean wind speeds are high (14 and 9 mph at the 200 and Satellite positions). The mean speed at the 50-foot location is surprisingly low and this situation is discussed in detail in section titled "Tabulation of Wind Directions and Speeds" below.

Turbulence Class III

As with Class I, the figures representing these cases (Figures 2.2-10 through 12) include a small percentage of the overall hours and are primarily associated with S to SE winds.

Turbulence Class IV

The Class IV roses (Figures 2.2-13, 14 and 15) also represent a small percentage of the hours, but they are more significant because of the relatively poor dispersion conditions associated with the classification. At all of the locations, the distributions are surprisingly uniform, with most direction sectors being approximately ½% of the hours and the peaks reaching 1 to 1½%.

All Classes

The last three figures of this series (Figures 2.2-16 through 18) include all hours and they are essentially similar to the Class II cases of which they are largely comprised.

Seasonal Variations

There is considerable seasonal variability in the wind roses, but nothing that is exceptionally significant from the point of view of dispersion problems. The overall wind rose data from the 200-foot level provides a typical picture of this variability. Figures 2.2-19 through 22 show the patterns for the four seasons. The winter rose (Figure 2.2-19) is quite varied with erratic peaks from several directions. The spring pattern is much more definite, Figure 2.2-20 showing a clear preference for NNW and SSE directions. The summer rose has the most pronounced pattern, with more than 6% in each of the sectors from 190 to 210 degrees. The fall wind rose has a tendency to reflect more easterly components than any of the others, with SE and NE prominently represented, as well as a single prominent peak from N.

Tabulation of Wind Directions and Speeds

The listing of wind speeds and directions by turbulence classes for direct use in the estimation of anual dose rates or concentration patterns from the small quantities of gaseous effluents released in normal operation is accomplished in Tables 2.2-11 through 18. The first four reflect the 200-foot wind distributions and the last four were derived from the inland satellite. The latter was used in preference to the 50-foot level on the main tower because of the restrictive influence of the dunes and vegetation on the flow through the lower levels of the tower. These tables are self-explanatory and simply document conclusions that are noted in other sections of this meteorology report.

Tables 2.2-19 through 28 are similar listings for the on-site beach instrument, arranged so that one can examine the complete patterns on a monthly basis, during each of the months that the equipment has been in operation so far. In these tables the Class IV hours are contrasted to the overall hourly data.

Representativeness of the Wind Speeds

After the original main tower installation and inland satellite had been in operation for the first full year, it became evident that there was some tendency for restriction of the low-level wind speeds as exemplified by the 50-foot level on the tower. In particular, it was noted that the mean winds at the 50-foot level were much lower than those at the inland satellite, which are actually closer to the ground surface. Table 2.2-29 shows the problem very clearly: the wind speeds at the 50-foot level are significantly lower than those at the satellite in all but the stable, Class IV turbulence. Furthermore, comparison of the 50-foot speeds with those obtained from the 200-foot instrument indicated an unreasonably rapid increase of wind speed with height, whereas comparison between the satellite and the 200-foot levels were more in accord with typical results.

This restriction apparently was associated with the vegetation nearby and with the rugged dune structure. Since the terrain was being altered locally for construction purposes, it was felt that a wind instrument located nearer the beach would be more representative and the 50-foot Aerovane on the inland satellite was moved to the location shown on the plot plan in Chapter 1 in the Spring of 1969. The instrument was replaced by a RAIM Associates cup and vane in December 1969 to provide greater sensitivity and accuracy at low speeds. The beach satellite is no longer in operation.

<u>Onshore Winds During Stable Conditions</u>

An important factor in safety analyses is the frequency of onshore winds accompanied by stable atmospheric conditions and the speed of such winds when such a condition occurs.

The data from the beach instrument are worth reviewing from this standpoint. In Table 2.2-30, the frequency of onshore winds associated with Class IV turbulence is presented for the five months in which the beach instrument has been operating satisfactorily. The data are further broken down according to wind speed in the 0-3 mph class and those exceeding 3 mph. "Onshore" is defined as any wind ranging from 180-010 degrees on the westerly side of the compass rose.

Except for the month of August, 1969, in which Class IV turbulence was common, the combination of onshore winds of low speeds in stable conditions is very unusual. Based on the data obtained so far, one would anticipate an annual freqency of occurrence of less than 1%.

Wind Steadiness

An analysis of the wind steadiness (its tendency to remain nearly fixed in direction over extended periods, as defined by Singer[5]) is presented in Table 2.2-31. Essentially, the tabulated data indicate how frequently one would anticipate nearly steady wind directions to occur for varying lengths of time. Considering any sort of dispersion condition (shown in the upper portion of the table) one would expect periods of 24 hours with almost invariant winds to occur about once a year, but steady winds lasting as long as 8 consecutive days should not occur more than once in 80 years. Of more direct significance to safety evaluation is the combination of steady winds and stable conditions (Class IV turbulence) given in the lower section. It appears that such conditions might occur for as long as 24 hours every three years or so, but two consecutive days of stable, steady winds would be most unlikely.

Dispersion Parameters

The bivane installation at 150 feet on the tower was originally planned as the primary indicator of the horizontal and vertical dispersion parameters. Unfortunately, as has been true in many similar installations, this piece of equipment is not adapted to routine service of any kind, especially in rigorous climates and it was in operation infrequently. Consequently, the dispersion parameters in Table 2.2-32 are derived partially from the Aerovane records of wind direction range and partially from general considerations of mid-latitude dispersion. The main contribution from the bivane was the suggestion that the vertical motion in the vicinity of the dunes is more vigorous than it would be over flat terrain and the expressions of δ_z in the stable case indicate this. The contribution of δ_y and δ_z for Class IV gives results very close to Pasquill F within the first kilometer from the source.

Atmospheric Pressure

One of the special studies completed during the course of the investigation is included in this report because of particular interest in the subject. This is the review of hourly changes in atmospheric pressure obtained from the aneroid that was included in the facility. Table 2.2-33 shows that the vast majority of the hours have net changes ranging from -.15 to +.15 inches, with a few scattered cases having larger rates of change. It is true that much higher rates of change may occur for a few minutes during an hour, but they do not persist.

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.

Turbulence Class III

As with Class I, the figures representing these cases (Figures 2.2-10 through 12) include a small percentage of the overall hours and are primarily associated with S to SE winds.

Turbulence Class IV

The Class IV roses (Figures 2.2-13, 14 and 15) also represent a small percentage of the hours, but they are more significant because of the relatively poor dispersion conditions associated with the classification. At all of the locations, the distributions are surprisingly uniform, with most direction sectors being approximately 1/2% of the hours and the peaks reaching 1 to 1 1/2%.

All Classes

The last three figures of this series (Figures 2.2-16 through 18) include all hours and they are essentially similar to the Class II cases of which they are largely comprised.

Seasonal Variations

There is considerable seasonal variability in the wind roses, but nothing that is exceptionally significant from the point of view of dispersion problems. The overall wind rose data from the 200-foot level provides a typical picture of this variability. Figures 2.2-19 through 22 show the patterns for the four seasons. The winter rose (Figure 2.2-19) is quite varied with erratic peaks from several directions. The spring pattern is much more definite, Figure 2.2-20 showing a clear preference for NNW and SSE directions. The summer rose has the most pronounced pattern, with more than 6% in each of the sectors from 190 to 210 degrees. The fall wind rose has a tendency to reflect more easterly components than any of the others, Wind Steadiness

An analysis of the wind steadiness (its tendency to remain nearly fixed in direction over extended periods, as defined by Singer5) is presented in Table 2.2-31. Essentially, the tabulated data indicate how frequently one would anticipate nearly steady wind directions to occur for varying lengths of time. Considering any sort of dispersion condition (shown in the upper portion of the table) one would expect periods of 24 hours with almost invariant winds to occur about once a year, but steady winds lasting as long as 8 consecutive days should not occur more than once in 80 years. Of more direct significance to safety evaluation is the combination of steady winds and stable conditions (Class IV turbulence) given in the lower section. It appears that such conditions might occur for as long as 24 hours every three years or so, but two consecutive days of stable, steady winds would be most unlikely.

Dispersion Parameters

The bivane installation at 150 feet on the tower was originally planned as the primary indicator of the horizontal and vertical dispersion parameters. Unfortunately, as has been true in many similar installations, this piece of equipment is not adapted to routine service of any kind, especially in rigorous climates and it was in operation infrequently. Consequently, the dispersion parameters in Table 2.2-32 are derived partially from the Aerovane records of wind direction range and partially from general considerations of mid-latitude dispersion. The main contribution from the bivane was the suggestion that the vertical motion in the vicinity of the dunes is more vigorous than it would be over flat terrain and the expressions of z in the stable case indicate this. The contribution of y and z for Class IV gives results very close to Pasquill F within the first kilometer from the source.

Turbulence Class I represents a very small percentage of the total observational period, as has been noted in Table 2.2-4, and it clearly is related to unstable lapse rates. Turbulence Class II (Table 2.2-7) is also primarily related to unstable lapse rates, but a significant portion of the cases are associated with stable lapse rates and strong wind speeds. The stormy conditions are represented by Class III, with most of the cases in Table 2.2-8 appearing in conjunction with high winds. The Class IV condition has an unusually wide scatter in its relation with both winds and lapse rates. Many, but by no means the majority, of the cases are found with inversions (Table 2.2-9) but light winds and instability also account for a number of Class IV types. This association is almost certainly a result of the flow direction from the lake, which can be lacking in turbulence if the winds are light and the lake surface reasonably calm.

Table 2.2-10 summarized the relation between lapse rates and wind speeds for all hours, regardless of turbulence class. The remarkable feature of the table is the wind distribution. Thirty-four percent of the winds at the 200-foot level exceed 18 mph, and only 3% fall into the 0-3 mph category. The lapse rate distribution, as has been noted earlier, is not especially unusual for a site with such strong winds. One notes some difference in the total percentages within lapse rate groups between Table 2.2-5 and Table 2.2-10. This results from the inclusion of all hours having acceptable lapse rate data in the former, whereas only those hours with both good wind and Lapse rate data are included in Table 2.2-10.

<u>Wind Direction and Speed Distributions</u>

There are many systems of summarizing wind data for a study of site dispersion characteristics, but the key questions are usually defined as the most probable annual dose rates from the small quantities of gaseous effluents released in normal operation and estimation of the least favorable conditions that might follow an accident. In the following with SE and NE prominently represented, as well as a single prominent peak from N.

Tabulation of Wind Directions and Speeds

The listing of wind speeds and directions by turbulence classes for direct use in the estimation of anual dose rates or concentration patterns from the small quantities of gaseous effluents released in normal operation is accomplished in Tables 2.2-11 through 18. The first four reflect the 200-foot wind distributions and the last four were derived from the inland satellite. The latter was used in preference to the 50-foot level on the main tower because of the restrictive influence of the dunes and vegetation on the flow through the lower levels of the tower. These tables are self-explanatory and simply document conclusions that are noted in other sections of this meteorology report.

Tables 2.2-19 through 28 are similar listings for the on-site beach instrument, arranged so that one can examine the complete patterns on a monthly basis, during each of the months that the equipment has been in operation so far. In these tables the Class IV hours are contrasted to the overall hourly data.

Representativeness of the Wind Speeds

After the original main tower installation and inland satellite had been in operation for the first full year, it became evident that there was some tendency for restriction of the low-level wind speeds as exemplified by the 50-foot level on the tower. In particular, it was noted that the mean winds at the 50-foot level were much lower than those at the inland satellite, which are actually closer to the ground surface. Table 2.2-29 shows the problem very clearly: the wind speeds at the 50-foot level are significantly lower than those at the satellite in all but the stable, Class IV turbulence. Furthermore, compari-

Sampling of Food

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

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-cow-milk pathway is critical and that milk is the best monitoring medium. Lake water is not used for irrigation in the area. There is, consequently, neither need nor justification for monitoring human foods other than milk in the terrestial environment, and fish in the aquatic environment.

All sampling points have been selected on their being representative of the area and accessible for sampling. Table 2.7-4 describes the current 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 possible monitor organisms 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.

TABLE 2.7-3

LOCATIONS OF THE MILK SAMPLING STATIONS

A) INDICATOR FARMS

BRIDGMAN:	DISTANCE: 4.25 MILE SECTOR: C Figure 2.7-2, FARM #M1
STEVENSVILLE:	DISTANCE: 4.5 MILE SECTOR: F Figure 2.7-2, FARM #M2
GALIEN:	DISTANCE: 9 MILE SECTOR: G Figure 2.7-2, FARM #M3

B) BACKGROUND FARMS

SOUTH BEND:	DISTANCE: 18 MILE SECTOR: E Figure 2.7-2, FARM #M4
DOWAGIAC:	DISTANCE: 20 MILE SECTOR: J Figure 2.7-2, FARM #M5

TABLE 2.7-4

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
1. Airborne a. Radioiodine and 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 Radioactivity following Filter Change[a], composite (by location) for gamma isotopic quarterly.
2. Direct Radiation	a) T1-T9 (Site) b) New Buffalo, South Bend, Dowagiac, Coloma c) 10 TLD Monitor Locations in the Five Mile Radius	At least once per 92 Days (Quarterly)	Gamma Dose. At Least Once Per 92 Days.
3. Waterborne a. Surface	L1, L2, L3	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.

TABLE 2.7-4 (Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
b. Ground	W1-W7	Quarterly	Gamma Isotopic and Tritium analysis quarterly.
c. Drinking	St. Joseph(D_B) Lake Township(D_A)	Composite* Sample Collected over a Period of ≤ 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	Semi-Annually	Gamma Isotopic Analysis Semi-Annually.
4. Ingestion a. Milk	Stevensville Bridgman Galien Dowagiac South Bend	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.

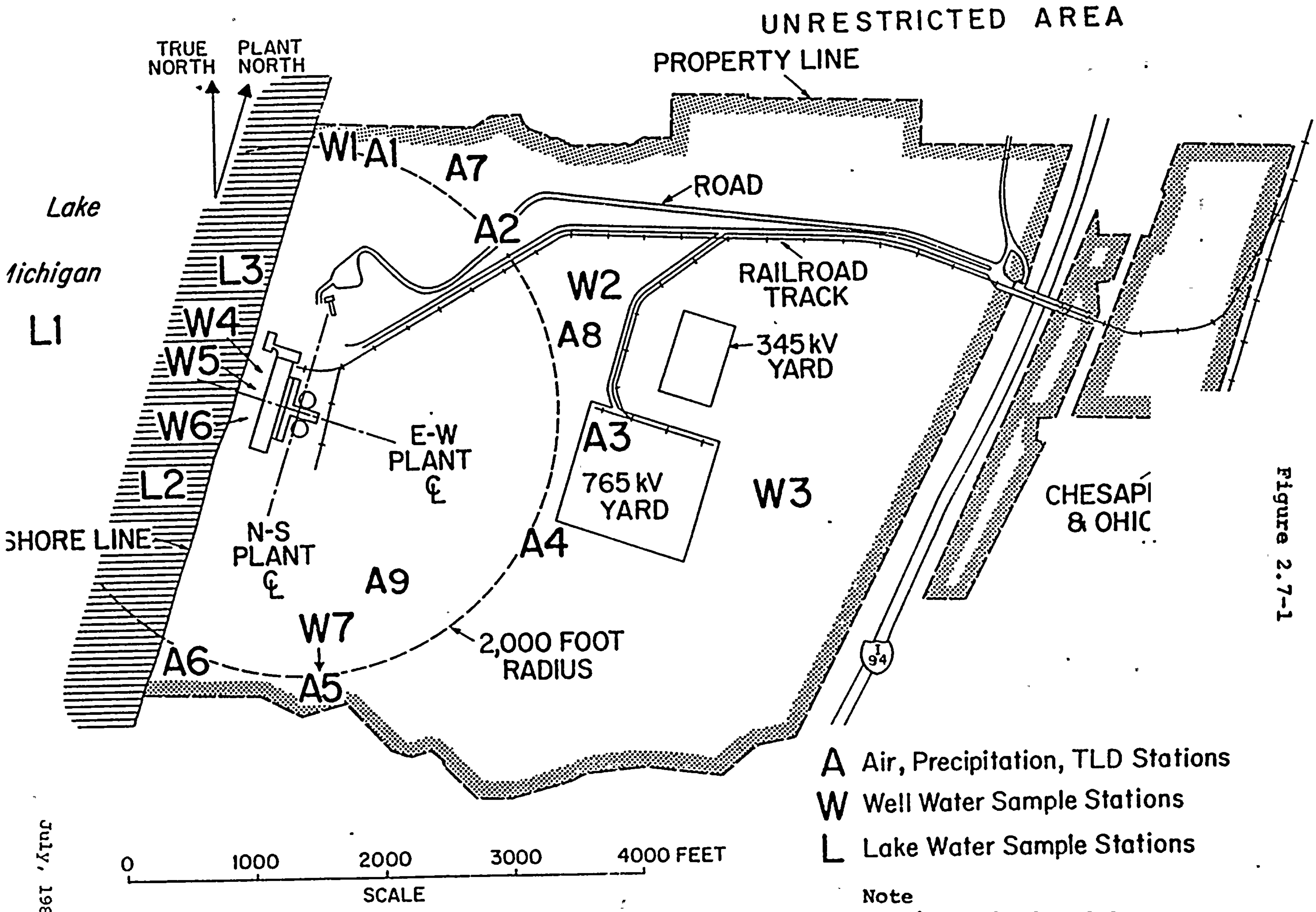
*Composite samples shall be collected by collecting an aliquot at intervals not exceeding 24 hours.

TABLE 2.7-4 (Cont'd)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Exposure Pathway And/Or Samples	Sample Locations	Sampling and Collection Frequency	Type & Frequency of Analysis
b. Fish	Plant Site Off-Site	2/year (Semi-Annually)	Gamma Isotopic Analysis on Edible Portion.
c. Food Products	Plant Site Off-Site (approx. 20 mi)	At time of Harvest. One Sample of Each of the Following Classes of Food Products: 1. Grapes	Gamma Isotopic Analysis on Edible Portion.
	Plant Site	At time of Harvest. One sample of Broad Leaf Vegetation	Gamma Isotopic Analysis

[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.



Note
Stations A7, 8 and 9 are TLD Stations Only

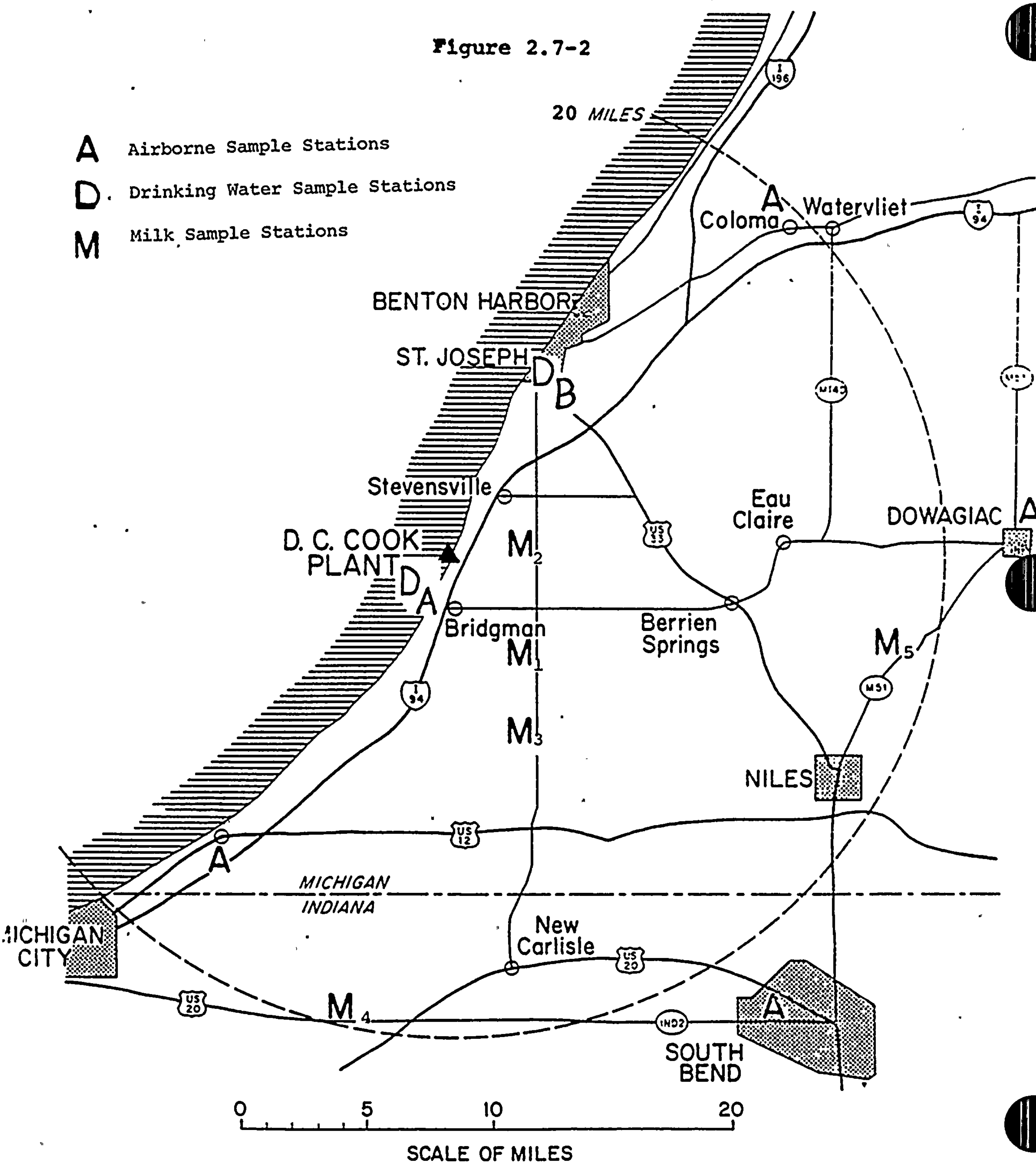
Figure 2.7-1

Figure 2.7-2

A Airborne Sample Stations

D Drinking Water Sample Stations

M Milk Sample Stations



In conclusion a set of "as built" dimensions were taken to verify conformance to the design requirements and assure proper fitup between the reactor internals and the reactor pressure vessel.

Fuel Quality Control

Quality Control philosophy is generally based on the following inspections being performed to a 95% confidence that at least 95% of the product meets specification, unless otherwise noted, using either a hypergeometric function with zero defects for small lots or the latest revision of Mil-105D for large lots. This confidence level has been based on past experience gained during the manufacturing of over 400 metric tons of uranium cores. The following inspections are included:

1) Component Parts

.All parts received are inspected to a 95/95 confidence level. The characteristics inspected depend upon the component parts and include dimensional and visual checks, audits of test reports, material certification and non-destructive testing such as X-ray and ultrasonic. Westinghouse materials process and component specifications specify in detail the inspection to be performed.

All material used in the manufacture of this core has been accepted and released by Westinghouse Quality Control.

2) Pellets

Inspection is performed to a 95/95 confidence level for the dimensional characteristics such as diameter, length and squareness of ends. Additional visual inspections are performed for cracks, chips and porosity according to standards established at the beginning of production. These standards are based upon standards used in previous cores which have in turn served as standards for over 50 million pellets manufactured and used in operating cores. Density is determined in terms of weight per unit length and is plotted on zone charts used in controlling the process. Chemical analyses are taken on a sample basis throughout pellet production.

3) Rod Inspection

Rod inspection consists of the following 100% non-destructive inspection and is based on the experience, specifications, procedures and standards established on previously manufactured and operating cores.

a) Leak Testing

Each rod is tested to a known leak rate using mass spectrometry with helium being the detectable gas. This is the system used previously on the leak test of over 300,000 rods.

b) X-ray

All fuel rod weld enclosures are X-rayed at 0°, 60°, and 120° using weld correction blocks. X-rays are taken in accordance with ASTM E-142-68, using 2-2T as the basis of acceptance.

c) Dimensional

All rods are dimensionally inspected prior to final release and upgrading. The requirements include such items as length, camber, and visual inspection.

This ensures that 100 percent of the rod welds have been checked by several different techniques.

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 UF_6 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 UF_6 to UO_2 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 UO_2 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.

Burnable Poison Rod Tests and Inspections

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 UO_2, 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".

artificially raised to the design value of $F^N_{\Delta H}$. Care is taken in the nuclear design of all fuel cycles and all operating conditions to ensure that a flatter assembly power distribution does not occur with limiting values of $F^N_{\Delta H}$.

Axial Power Distributions

The shape of the power profile in the axial or vertical direction is largely under the control of the operator through manual and automatic motion of full length rods, and by responding to manual operation of the CVCS. Nuclear effects which cause variations in the axial power shape include moderator density, Doppler effect on resonance absorption, spatial xenon, and burnup. Automatically controlled variations in total power output and full length rod motion are also important in determining the axial power shape at any time. Signals are available to the operator from the excore ion chambers which are long ion chambers outside the reactor vessel running parallel to the axis of the core. Separate signals are taken from the top and bottom halves of the chambers. The difference between top and bottom signals from each pair of detectors is displayed on the control panel and called the flux difference, ΔI. Calculations of the core average peaking factor for many plants and measurements from operating plants under many operating situations are associated with either ΔI or axial offset in such a way that an upper bound can be placed on the peaking factor. For these correlations, axial offset is defined as:

$$\text{Axial offset} = \frac{\phi_t - \phi_b}{\phi_t + \phi_b}$$

where ϕ_t and ϕ_b are the top and bottom detector readings.

Representative axial power shapes for BOL and EOL conditions are shown in Figure 3.3.1-16.

Limiting Power Distributions

Occurrences which are expected frequently or regularly in the course of power operation, maintenance, or maneuvering of the plant are accommodated with margin between any plant parameter and the value of that parameter which would require either automatic or manual protective action. Inasmuch as these occurrences occur frequently or regularly, they must be considered from the point of view of affecting the consequences of fault conditions. In this regard, analysis of each fault condition described is generally based on a conservative set of initial conditions corresponding to the most adverse set of conditions which can occur during normal operations.

The list of steady-state and shutdown conditions, permissible deviaticns (such as one coolant loop out of service) and operational transients is given in Chapter 14. Implicit in the definition of normal operation is proper and timely action by the reactor operator. That is, the operator follows recommended operating procedures for maintaining appropriate power distributions and takes any necessary remedial actions when alerted to do so by the plant instrumentation. Thus, as stated above, the worst or limiting power distribution which can occur during normal operation is to be considered as the starting point for analysis of fault conditions.

Improper procedural actions or errors by the operator are assumed in the design as occurrences of moderate frequency. Some of the consequences which might result are discussed in Chapter 14. Therefore, the limiting power shapes which result from such events, are those power shapes which deviate from the normal operating condition at the recommended axial offset band, e.g., due to lack of proper action by the operator during a xenon transient following a change in power level brought about by control rod motion. Power shapes which fall in this category are used for determination of the Reactor Protection System set points so as to maintain margin to overpower the DNB limits.

Doppler and Power Coefficients

The Doppler coefficient is defined as the change in neutron multiplication per degree change in fuel temperature. The coefficient is obtained by calculating neutron multiplication as a function of effective fuel temperature[3]. The results from initial calculations are shown in Figure 3.3.1-16.

In order to know the change in reactivity with power, it is necessary to know the change in the effective fuel temperature with power, as well as the Doppler coefficient. It is very difficult to predict the effective temperature of the fuel using a conventional heat transfer model because of uncertainties in predicting the behavior of the fuel pellets. Therefore, an empirical approach is taken to calculate the power coefficient, based on operating experience of existing Westinghouse fueled cores. Figure 3.3.1-17 shows the power coefficient as a function of power obtained by this method. The results presented do not include any moderator coefficient even though the moderator temperature changes with power level.

Nuclear Evaluation

The basis for confidence in the procedures and design methods comes from the comparison of these methods with many experimental results. These experiments include criticals performed at the Westinghouse Reactor Evaluation Center (WREC) and other facilities, and also measured data from operating power reactors. A summary of the results and discussion of the agreement between calculated and measured values is given in other Safety Analysis Reports such as the FSAR for Indian Point Unit 2, Docket No. 50-247, Section 3.2.1, and the PSAR for D. C. Cook, Docket No. 50-315-316, Section 3.2.1.

Extensive analyses on the threshold to xenon instabilities as a function of variation in core parameters (power coefficient, etc.) have been reported in Reference 4.

Finally, verification of design analysis during the startup physics tests is described in Section 3.3.2.

3.3.2 PHYSICS TESTS

Tests to Confirm Reactor Core Characteristics

A detailed series of startup physics tests were performed from zero power up to and including 100% power. As part of these tests, a series of core power distribution measurements were made over the entire range of operation in terms of RCCA configuration and power level by means of the incore movable detector system. In addition, rod worth, born end-point, and reactivity coefficient measurements were made.

Within relevant acceptance criteria, these test results show good agreement with design predictions[1]. To detect and eliminate possible errors in the calculations of the initial reactivity of the core and the reactivity depletion rate, the predicted relationship between fuel burnup and the boron concentration was normalized to accurately reflect actual core conditions. When full power was initially reached, and with the control groups in the desired positions, the boron concentration was measured and the predicted curve was adjusted to this point. As power operation continued, the measured boron concentration was compared with the predicted concentration and the slope of the predicted curve relating burnup and reactivity was corrected as necessary. This normalization was completed after about 10 percent of the total core burnup has occurred. Thereafter, actual boron concentration was compared with the predicted concentration, and the reactivity prediction of the core was continuously evaluated. No reactivity anomaly greater than one percent was observed.

In the THINC analysis, the benefit of coolant mixing in all the subchannels in the hot assembly is considered and a mixing factor of approximately 0.90 is used to evaluate the enthalpy rise to the point of minimum DNB ratio.

The above subfactors are combined to obtain the total engineering hot channel factor for an enthalpy rise of 1.01. The reduction in this subfactor at nominal operating conditions from a value of 1.075 (PSAR) was the result of the adaption of the THINC code (multi-subchannel analyses) as a thermal and hydraulic design method. Table 3.4.1-2 is a tabulation of the design engineering hot channel factors.

Operational Limits:

The above subfactors are incorporated in THINC steady-state and transient analyses to yield operating limits for the maximum measured value of the enthalpy rise hot channel factor, $F_{\Delta H}^{N}$ For the D. C. Cook Plant Unit 1 the technical specification limit (for Cycle 7) is:

$$F_{\Delta H}^{N} = 1.51 \; [1 + 0.2 \; (1-P)]^{*} \qquad (2a)$$

where, P is the ratio of operating power to rated power. The engineering subfactor $F_{\Delta H}^{E}$ is incorporated into the limiting value of 1.51*, and

$$F_{\Delta H}^{N} = 1.04 \; F_{H}^{N'} \qquad (2b)$$

where, $F_{\Delta H}^{N'}$ is the measured nuclear enthalpy rise peaking factor, and the factor of 1.04 accounts for measurement uncertainty.

* For Cycles 8 and 9, the $F_{\Delta H}^{N}$ limiting values for Westinghouse and ENC fuel at 3250 MWt rated power have been changed to the values stated in Tables 3.3.1-1 and 3.6.3-1.

The heat flux engineering subfactor of 1.03 is included in the maximum measured value of the heat flux hot channel factor,

$$F_Q^N = 1.03 \times 1.05\ F_Q^{N'} \qquad (2c)$$

where, F_Q^N is the measured nuclear hot channel factor and the factor of 1.05 accounts for measurement uncertainties. For Cycle 9 operations, the technical specifications require that F_Q^N not exceed the limits defined in Section 3.2.2 of the technical specifications. (See Section 3.6.2.2).

Pressure Drop and Hydraulic Forces

The total loss across the reactor vessel, including the inlet and outlet nozzles, and the pressure drop across the core are listed in Table 3.4.1-1. These values include a 10% uncertainty factor.

Thermal and Hydraulic Design Parameters

The thermal and hydraulic design parameters are given in Table 3.4.1-1.

Thermal and Hydraulic Evaluation

W-3 Equivalent Uniform Flux DNB Correlation

The equivalent uniform DNB flux $q''_{DNB,EU}$ is calculated from the W-3 equivalent uniform flux DNB correlation as follows:

$$\frac{q''_{DNB,EU}}{10^6} = [(2.022 - 0.0004302p) + (0.1722 - 0.0000984p)e^{(18.177 - 0.004129p)\chi}]$$

$$\times\ [1.037 + \frac{G}{10^6}(0.1484 - 1.596 + 0.1729\chi|\chi|)] \times [1.157 - 0.869\chi]$$

$$\times\ [0.2664 + 0.8357e^{-3.151D_e}] \times [0.8258 + 0.000794\,(H_{sat} - H_{in})] \qquad (3)$$

The two dominant considerations in the Cycle 2 core design were: 1) maintaining the low F_Q^N value, and, 2) have a zero or negative moderator temperature coefficient (MTC) at BOC. The Cycle 2 core design was intended to establish and maintain a relatively flat power distribution to ensure a low nuclear peaking factor. The negative MTC was assured by administratively keeping the boron concentration below the value where the MTC turns positive.

The reference core loading, which assumed a burnup of 17,200 MWD/MT in Cycle 1, included 96 burnable poison rods distributed in 16 Exxon Nuclear assemblies. Sufficient calculations were performed for the reference core to determine that the neutronic parameters were adequate for utilization in the accident and safety analysis. Nominal values of the neutronic parameters for the Cycle 2 core lie within the ranges analyzed in the FSAR for Cycle 1, and within the bounding values used in the safety analysis for Cycle 2. The MTC of reactivity for the Cycle 2 core was in the range from 0.0×10^{-4} to -3.0×10^{-4} $\Delta\rho/°F$. Similarly, the bounding values of the MTC used in the safety analysis for Cycle 2 were 0.0×10^{-4} at BOC and -3.2×10^{-4} $\Delta\rho/°F$ at EOC, while comparable Cycle 1 FSAR values are 0.0×10^{-4} and -3.5×10^{-4} $\Delta\rho/°F$, respectively.

Physics Characteristics

The neutronic characteristics of the Cycle 2 core are compared with those of Cycle 1 in Table 3.5.2-1. The Cycle 2 reference reactivity coefficients are bounded by those of the safety analysis.

The Cycle 2 limit on the total power peaking factor (F_Q) of 1.98, which allows for a calculational uncertainty of 5 percent and 3 percent for engineering factors, was accommodated. This corresponds to a F_Q^N limit of 1.83. This limit assured that the peak fuel rod linear power density remained below the limiting values, thus meeting the LOCA and DNBR overpower limits criteria.

The worth of all control rods inserted in Cycle 2 was comparable to the worths seen in Cycle 1, thus, indicating comparable shutdown margins.

The control rod grouping and insertion sequence for Cycle 2 was not changed from that of Cycle 1.

Analytical Input

The neutronics design methods utilized to calculate the data presented herein are consistent with those described in Reference 1 with primary reliance upon the XTG simulator code, Reference 2.

The burnup history of each of the exposed fuel assemblies was calculated by a three-dimensional, four node per assembly XTG model which was utilized to simulate the Cycle 1 operation of the core. The results of this calculational model are compared to a core measured power distribution in Figure 3.5.2-1 and the boron curve in Figure 3.5.2-2.

Calculations for BØC2 utilized the assembly exposures, four values per assembly, calculated in Cycle 1 at 17,200 MWD/MTU. The 3-D XTG model was verified using the 2-D pin-by-pin PDQ 7/HARMONY model. Axial effects were accounted for through the buckling term B_z^2.

<u>Unit 1, Cycle 5 and Cycle 6 Neutronic Design</u>

D.C. Cook Unit 1 Cycle 5 was chosen as the reference cycle with respect to Cycle 6 due to close resemblence of the neutronic characterisitics between these two cycles. The end of cycle exposure was 10,653 MWD/T. The Cycle 5 and 6 consisted exclusively of assemblies supplied by ENC.

Design Basis

The Cycle 6 loading pattern was designed to achieve power distributions and control rod reactivity worths according to the following constraints:

a. The peak F_Q is not to exceed the exposure dependent limit shown in Figure 3.3.1-17, and the peak $F_{\Delta H}$ is not to exceed 1.51.

b. The scram worth of all rods, minus the most reactive rod, is to be greater than the BOC and EOC shutdown requirements.

c. The moderator temperature coefficient is not to exceed +5 pcm/°F at HZP, and less than 0 pcm/°F at 70% RTP.

Physics Characteristics

The neutronic characteristics of the Cycle 6 core are compared to those of the Cycle 5 core in Table 3.5.2-2. The data indicate the neutronic similarity between Cycles 5 and 6. Cycle 6 calculations of radial power distributions at BOL, MOL, and EOL HFP conditions are shown in Figures 3.5.2-3 through 3.5.2-6. The boron letdown curve for Cycle 6 is shown in Figure 3.5.2-7, and a comparison between predicted power distribution (XTG) and measured power distribution (Flux Map 106-15) is shown in Figure 3.5.2-8. The V(Z) factor, which is used to obtain the maximum anticipated $F_Q^T(Z)$ max for Cycles 2-7, is shown in Figure 3.5.2-9.

REFERENCES, SECTION 3.5.2

1. Exxon Nuclear Neutronic Design Methods for Pressurized Water Reactors, XN-75-27 and XN-75-27, Rev. 1

2. XTG, A Two-Group Three-Dimensional Reactor Simulator Using Coarse Mesh Spacing, XN-CC-28, Rev. 3.

3. Axial Power Distribution Mooonitoring System Experience and Peaking Factor Determination at the Donald C. Cook Nuclear Plant Plant Unit No. 1, Oct. 15, 1975 and Addendum No. 1, Dec. 31, 1975, American Electric Power.

design. The bases and criteria given in Section 3.2.1.1.1 of the Cook Unit 2 FSAR 1983 update[10] are also applicable, but it should be noted that the region average discharge burnups considered in the Cook Unit 1 OFA fuel design are typically in the range of 38,000 MWD/MTU. These design bases and criteria are summarized below:

a. The cladding stresses under Condition I and II events are less than the Zircaloy 0.2% offset yield stress, with due consideration of temperature and irradiation effects. While the cladding has some capability for accommodating plastic strain, the yield stress has been accepted as a conservative design basis.

b. Cladding Tensile Strain - The total tensile creep strain is less than 1% from the unirradiated condition. The elastic tensile strain during a transient is less than 1% from the pre-transient value. This limit is consistent with proven practice.

c. Strain Fatigue - The cumulative strain fatigue cycles are less than the design strain fatigue life. This basis is consistent with proven practice.

d. Wear - Potential for fretting wear of the clad surface exists due to flow induced vibrations. This condition is taken into account in the design of the fuel rod support system. The clad wear depth is limited to acceptable values by the grid support dimple and spring design.

e. 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.[11]

   Rod pressure is also limited such that extensive DNB propagation shall not occur during normal operation and accident events.[11]

f. Cladding collapse shall be precluded during the fuel rod design lifetime. The models described in Reference 12 are used for this evaluation.

g. During modes of operation associated with Condition I and Condition II events, there is at least a 95 percent probability that the peak kW/ft fuel rods will not exceed the UO_2 melting temperature. The melting temperature of UO_2 is taken at 5080°F[7], unirradiated and decreasing 58°F per 10,000 MWD/MTU. By precluding UO_2 melting, the fuel geometry is preserved and possible adverse effects of molten UO_2 on the cladding are eliminated. To preclude center melting, and as a basis for overpower protection system setpoints, a calculated centerline fuel temperature of 4700°F has been selected as the overpower limit.

h. Design values for the properties of materials used for the fuel rod design and performance are given in Reference 7.

Evaluation

The detailed OFA fuel rod design establishes such parameters as pellet size and density, cladding-pellet diametral gap, gas plenum size, and helium pre-pressurization level. The design also considers effects such as fuel density changes, fission gas release, cladding creep, and other physical properties which vary with burnup. The integrity of the fuel rods is ensured by designing to prevent excessive fuel temperatures, excessive internal rod gas pressures due to fission gas releases, and excessive cladding stresses and strains. This is achieved by designing the fuel rods to satisfy the conservative design bases in the following subsections during Condition I and Condition II events over the fuel lifetime. For each design basis, the performance of the limiting fuel rod must not exceed the limits specified. The NRC approved fuel rod design model[13,14] is used to assure that design bases are satisfied and to predict fuel operating characteristics. Additional details in the evaluation of the OFA fuel rods, which show that the design bases are satisfied, are given in Sections 4.2.3.1, 4.2.3.2 and 4.2.3.3 of WCAP-9500[6]. Also applicable are the fuel rod evaluations given in Section 3.2.1.3.1 of the Cook Unit 2 FSAR 1983 update[10].

The W 15x15 OFA fuel rod design is essentially the same as the LOPAR W 15x15 fuel rod design which has exhibited good in-core fuel performance[2]. The W OFA and ENC fuel rods have similar length and clad OD dimensions. Table 3.6.1 presents a comparison of the W and ENC fuel rod designs.

As stated in the 17x17 OFA Reference Core Report[6], for a given burnup, the magnitude of rod bow for the W OFA is conservatively assumed to be the same as that of a W LOPAR fuel assembly. The most probable causes of significant rod bow are rod-grid and pellet-clad interaction forces and wall thickness variation. Since the OFA fuel rods are the same as the W LOPAR fuel rods, there will be no difference in predicted bow due to rod considerations. The OFA design will have reduced grid forces due to the Zircaloy grid springs. Therefore, this component is predicted to decrease OFA rod bow compared to LOPAR fuel. The impact of rod bow on DNBR penalties is discussed in Section 3.6.3.

The wear of fuel rod cladding is dependent on both the support provided by the grids and the flow environment to which it is subjected. OFA and ENC assembly flow test results were evaluated. ENC hydraulic test results show that the crossflow between ENC and W 15x15 LOPAR assemblies is very similar to that obtained during W flow tests on side-by-side W 15x15 OFA and W 15x15 LOPAR assemblies. These tests showed only a small crossflow between assemblies and no significant fuel rod wear due to rod vibration. Extrapolation of the results from flow tests involving OFA and LOPAR assemblies shows that fuel rod wear would be less than ten (10) percent of the cladding thickness for at least 48 months of reactor operation. This assures that clad wear will not impair fuel rod integrity.

The above conclusions on OFA rod wear and integrity have also been supported by analytical results. The analysis accounted for rod vibrations caused by both axial and crossflows, and for the effect of potential fuel rod to grid gaps.

3.6.1.4 Core Components

The core components consist of the rod cluster control assemblies (RCCAs), the primary and secondary source assemblies, the thimble plug assemblies, and the burnable absorber assemblies. The control rod assemblies in the Cook Unit 1 core are unchanged from previous cycles and are compatible with the OFA guide thimbles. New secondary source assemblies and OFA compatible plugging devices were supplied in Cycle 8. As discussed in Section 3.6.1.2.1, the reduced diametral clearance compared to ENC guide thimble results in an increased RCCA scram time from 1.8 to 2.4 seconds which is used in all accident reanalyses.

The guide thimble plug used with the OFA has a smaller diameter (0.485") than the current thimble plug diameter (0.498"), in order to maintain the same thimble plug to thimble tube diametral clearance. The thimble plug assembly presently used in ENC fuel cannot be used in OFAs due to insufficient diametral clearance between the current thimble plug and OFA guide thimble tube.

The optimized assemblies, their thimble plugging devices, and source assemblies are compatible with existing handling tools. A new tool is provided for handling the new Wet Annular Burnable Absorber (WABA) rods.

Wet Annular Burnable Absorber (WABA)

The Wet Annular Burnable Absorber (WABA) rod design will be used in the Cook Unit 1 reload cores with 15x15 W OFA fuel. The materials, mechanical, thermal hydraulic, and nuclear design evaluations of the WABA rods are presented in a topical report[5], which has received NRC generic approval[5] and approval for Cook Unit 1 application[1] of WABAs.

The WABA design has annular aluminum oxide - boron carbide (Al_2O_3 - B_4C) absorber pellets contained within two concentric Zircaloy tubes with water flowing through the center tube as well as around the outer tube. The WABA design provides significantly enhanced nuclear characteristics, when compared with the W borosilicate absorber rod design. Fuel cycle benefits result from the reduced parasitic neutron absorption of Zircaloy compared to stainless steel tubes, increased water fraction in the burnable absorber cell, and a reduced boron penalty at the end of each cycle.

Figures 3.6.1-8 and 3.6.1-9 show the design of a WABA rod, and Table 3.6.1-2 and Figure 3.6.1-9 present a comparison between the WABA rod and a W borosilicate glass absorber rod.

APOLLO is used to calculate differential rod worth versus core height, axial nuclear hot channel factor versus control rod height, axial xenon oscillations and stability studies, and axial power distributions.

3.6.2.2 Unit 1 Cycle 9 Neutronic Design

3.6.2.2.1 Analytical Input

The neutronics design methods utilized to calculate the data presented herein are consistent with those described in References 3-7 with primary reliance upon the 3D PALADON code.

The burnup history of each of the exposed fuel assemblies was calculated by a three-dimensional, four node radially and 48 node axially per assembly, 3D-PALADON model which was utilized to simulate operation of the core for Cycles 5, 6 and 7.

Calculations for BOC 9 utilized the assembly exposures, calculated at an EOC 7 burnup of 10,446 MWD/MTU, and at the EOC8 burnup of 15,681 MWD/MTU. The 3D PALADON model was verified using the 2D pin-by-pin TURTLE model. Axial effects in the 2D models are accounted for through the bucking term B_z^2.

3.6.2.2.2 Design Bases

The nuclear design bases for the Cycle 9 core are as follows:

1. The design shall permit operation within the Technical Specifications for the D. C. Cook Unit 1 nuclear plant.

2. The Cycle 9 loading pattern shall permit full power (3250 MWt total power) operation of the core throughout the Cycle 9 reactivity life time of about 15,750 MWD/MTU. Power distributions and control rod worth (both shutdown worth and the worth of a potentially ejected rod) are maintained within the ranges analyzed in the Cycle 8 safety analysis.

3. At hot full power (3250 MWt total power) the peak $F^N_{\Delta H}$ shall not exceed 1.435 in any single fuel rod throughout the cycle under nominal operating conditions.

4. The moderator temperature coefficient (MTC) is maintained less than or equal to +5 pcm/$^{\circ}$F below 70% of rated power and less than or equal to 0 pcm/$^{\circ}$F above 70% of rated power.

5. The worth of all rods minus the most reactive stuck rod shall exceed BOC and EOC shutdown requirements.

3.6.2.2.3 Design Description and Results

The Cycle 9 reactor core consists of a mixed W OFA/ENC fuel core of 193 assemblies, each having a 15x15 fuel rod array. A description of the W OFAs and ENC fuel assemblies are given in Sections 3.6.1 and 3.5.1 respectively.

The Cycle 9 loading pattern is given in Figure 3.6.2-1 which shows the region number, sources, and the burnable absorber configuration. The core consists of 48 fresh W OFAs with an average enrichment of 3.4 w/o U-235, 32 fresh OFAs with an average enrichment of 3.6 w/o, 79 once burnt OFA assemblies and 34 exposed ENC assemblies. A low leakage loading pattern was developed which results in the scatter-loading of the OFAs throughout the interior of the core. WABA rods are inserted into a number of OFAs to control power peaking and MTC. The exposed ENC fuel is also scatter-loaded in the center in a manner to control the power peaking. The WABA rods contain 0.0153 gm/in of B-10, and 576 of these rods are distributed among 56 fresh assemblies loaded in the core interior. Pertinent fuel assembly parameters for the Cycle 9 fuel are given in Tables 3.6.1-1 and 3.6.2-1.

<u>Physics Characteristics</u>

The nuetronics characteristics of the Cycle 9 core are compared with those of Cycle 8 and are presented in Table 3.6.2-2. The reactivity coefficients of the Cycle 9 core are bounded by the coefficients used in the safety analysis.

TABLE 3.6.2-2

D. C. Cook Unit 1 Neutronics Characteristics of
Cycle 9 Compared with Cycle 8 Data

	Cycle 8		Cycle 9	
	BOC	EOC	BOC	EOC
Critical Boron				
HFP, ARO, Equilibrium Xenon (ppm)	1098	10	1782	10
HZP, ARO, No Xenon (ppm)	1534	---	1313	---
Moderator Temperature Coefficient				
HFP, ARO (pcm/^{o}F)	-5.38	-25.65	-6.30	-25.65
HZP, ARO (pcm/^{o}F)	+2.82	---	+2.53	---
Doppler Coefficient (pcm/^{o}F)	-2.11	-2.23	-2.54	-2.62
Boron Worth, HZP (pcm/ppm)	-9.1	-11.4	-8.20	-10.10
Total Nuclear Peaking Factor				
F_Q^N, HFP, Equilibrium Xenon	1.596	1.586	1.688	1.527
Delayed Neutron Fraction	0.0061	0.0052	0.0060	0.0051
Control Rod Worth of All Rods in				
Minum Most Reactive Rod, HZP (pcm)	6060	6520	6088	6749
Excess Shutdown Margin (pcm)	1020	880	975	1100

TABLE 3.6.2-3

D.C. Cook Unit 1 Control Rod Shutdown
Margins and Requirements of
Cycle 9 Compared to Cycle 8

	Cycle 8		Cycle 9	
	BOC	EOC	BOC	EOC
Control Rod Worth (HZP), pcm				
All Rods Inserted (ARI)	7422	7849	6955	7550
ARI less most reactive (N-1)	6060	6520	6088	6749
N-1 less 10% allowance [(N-1)*.9)]	5450	5870	5479	6074
Reactivity Insertion, pcm				
Power Defect (Moderator + Doppler)	1180	1870	1355	1954
Flux Redistribution	410	970	500	870
Void	50	50	50	50
Sum of the above three	1640	2890	1905	2874
Rod insertion allowance	1190	500	999	500
Total Requirements	2830	3390	2904	3374
Shutdown Margin				
(N-1) * .9 - Total Requirements	2620	2480	2575	2700
Required Shutdown Margin	1600(a)	1600(a)	1600(a)	1600(a)
Excess Shutdown Margin	1020	880	975	1100

(a) Technical Specification Limit for Cycle 8 and 9

TABLE 3.1-1 (cont'd.)

REACTOR DESIGN COMPARISON TABLE

CORE MECHANICAL DESIGN PARAMETERS	D.C. COOK UNIT 2.	TROJAN
STRUCTURE CHARACTERISTICS		
51. Core Diameter, in (Equivalent)	132.7	132.7
52. Core Height, in (Active Fuel)	143.7	143.7
REFLECTOR THICKNESS AND COMPOSITION		
53. Top - Water plus Steel, in	10	10
54. Bottom - Water plus Steel, in	10	10
55. Side - Water plus Steel, in	15	15
56. H_2O/U Molecular Ratio Core, Lattice (Cold)	2.41	2.43
FEED ENRICHMENT, W/O, INITIAL CORE[h]		
57. Region 1	2.10	2.10
58. Region 2	2.60	2.60
59. Region 3	3.10	3.10

NOTES FOR TABLE 3.1-1

[a] These numbers are based on Improved Thermal Design Procedure in Reference 2.

[b] The value of 437,8000 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 limit is associated with the value of $F_Q = 3.50$

[d] 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.

[e] See Section 3.3.2.2.6.

[f] 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.

[g] Includes the effect of fuel densification.

[h] The reload feed enrichments for Cycle 2 and Cycle 3 were 3.4 w/o.

TABLE 3.3-1

REACTOR CORE DESCRIPTION

(First Cycle)

Active Core	
Equivalent Diameter, in	132.7
Active Fuel Height, First Core, in	143.7
Height-to-Diameter Ratio	1.08
Total Cross Section Area, ft^2	96.06
H_2O/U Molecular Ratio, lattice (Cold)	2.41
Reflector Thickness and Composition	
Top - Water plus Steel, in	10
Bottom - Water plus Steel, in	10
Side - Water plus Steel, in	15
Fuel Assemblies	
Number	193
Rod Array	17 x 17
Rods per Assembly	264
Rod Pitch, in	0.496
Overall Transverse Dimensions, in	8.426 x 8.426
Fuel Weight (as UO_2), lb	222,739
Zircaloy Weight, lb	50,913
Number of Grids per Assembly	8 Type R
Composition of Grids	INC718
Weight of Grids (Effective in Core), lb	2324
Number of Guide Thimbles per Assembly	24
Composition of Guide Thimbles	Zircaloy 4
Diameter of Guide Thimbles (upper part), in	0.450 I.D. x 0.482 O.D.
Diameter of Guide Thimbles (lower part), in	0.397 I.D. x 0.429 O.D.
Diameter of Instrument Guide Thimbles, in	0.450 I.D. x 0.482 O.D.

TABLE 3.3-1 (Continued)

REACTOR CORE DESCRIPTION

(First Cycle)

Fuel Rods	
Number	50,952
Outside Diameter, in	0.374
Diameter Gap, in	0.0065
Clad Thickness, in	0.0225
Clad Material	Zircaloy-4
Fuel Pellets	
Material	UO_2 Sintered
Density (percent of Theoretical)	95
Fuel Enrichments w/o	
Region 1	2.10
Region 2	2.60
Region 3	3.10
Diameter, in	0.3225
Length, in	0.530
Mass of UO_2 per Foot of Fuel Rod, lb/ft	0.364
Rod Cluster Control Assemblies	
Neutron Absorber	Ag-In-Cd
Composition	80%, 15%, 5%
Diameter, in	0.341
Density, lb/in^3	0.367
Cladding Material	Type 304, Cold Worked Stainless Steel
Clad Thickness, in	0.0185
Number of Clusters	
Full Length	53
Part Length	0
Number of Absorber Rods per cluster	24
Full Length Assembly Weight (dry), lb	157

3.5 EXXON FUEL DESIGN

The Exxon Nuclear Company (ENC) reload fuel assemblies described in this chapter are used in the Donald C. Cook Nuclear Plant Unit 2 Cycle 5, the cycle now in operation. This chapter describes the mechanical, nuclear, and thermal hydraulic design of these ENC-fabricated assemblies.

Cycle 4 operation of Donald C. Cook Nuclear Plant Unit 2 was the initial insertion of 17x17 fuel fabricated by ENC into a Westinghouse reactor. In addition, ENC assemblies differ slightly from the co-resident Westinghouse assemblies in fuel rod dimensions and water-to-fuel ratio, having been optimized for the higher burnup evident in the safety analyses.

The design of the ENC 17x17 array fuel assemblies is similar to the ENC 14x14 and 15x15 array fuel assemblies used successfully in other reactor applications as shown in Figures 3.5.1-1 and 3.5.1-2. Overall length, envelopes, spacer design, plenum length, etc., are identical. All dimensions affecting the mechanical interfacing with control rods and core support structure and with the co-resident fuel were maintained identical with the corresponding dimensions in the Westinghouse fuel. The methodology used by ENC in determining the appropriate operating setpoints and in verifying the safety of the reload core is described in a number of Licensing Topical Reports issued by ENC. A bibliography of the applicable topical reports is included in the reference sections of this chapter.

The insertion of ENC fuel into the core of Donald C. Cook Nuclear Plant Unit 2 involved the use of the NRC-approved XNB Departure from Nucleate Boiling (DNB) correlation developed by ENC and the NRC-approved ENC fuel densification model for PWRs. Operation of the plant is based on the use of ENC's NRC-approved Power Distribution Control Phase II (PDC-II), which is described earlier in Section 3.3 of this document.

3.5.1 FUEL AND MECHANICAL DESIGN

This section describes the mechanical, chemical and thermal design for the Donald C. Cook Nuclear Plant Unit 2 reload fuel in Batches 6 and 7 under normal operating conditions.

Exxon Nuclear's design configuration for the Donald C. Cook Nuclear Plant Unit 2 reload assemblies is compatible with Westinghouse fuel and Westinghouse designed reactor internals and consists of a 17x17 square array of 289 positions, occupied by 264 fuel rods, 24 Zircaloy-4 guide tubes and one Zircaloy-4 instrumentation tube.

The fuel consists of pressed and sintered UO_2 pellets. The nominal pellet density is 94.0% of the theoretical density, each pellet is dished on each end, and the fuel active length is nominally 144 inches. Zircaloy-4 end caps are seal welded to the Zircaloy-4 cladding. Fuel rod pitch is maintained by eight bi-metallic grid spacers constructed of Zicaloy-4 structural members with Inconel springs. The grids are equally spaced along the length of the fuel bundle and are welded to the guide tubes. The Zircaloy-4 guide tubes are mechanically attached and secured to the upper and lower tie plates. The spacers, guide tubes and tie plates form the structural skeleton of the fuel bundle. The upper tie plate is designed to be mechanically dismountable by remote handling under water.

Description of the Donald C. Cook Nuclear Plant Unit 2 Batches 6 and 7 assembly components, including purpose and rationale, is given in Table 3.5.1-1.

Mechanical Design

Fuel Assembly

Design Basis

The fuel assembly shall be dimensionally and hydraulically compatible with existing fuel and dimensionally compatible with reactor fuel handling equipment.

Design Evaluation

The significant dimensional comparisons between Exxon Nuclear and Westinghouse fuel designs are shown in Table 3.5.1-2. Exxon Nuclear has maintained the same dimensional values of all critical items. The only significant difference is clad diameter and thickness where Exxon Nuclear fuel rods are 4% smaller in diameter but have 10% thicker cladding than Westinghouse fuel rods.

Orientation of each fuel assembly is controlled by an assymmetric index hole in the upper tie plate. The position and size of the indexing hole with respect to the locating holes in Exxon Nuclear fuel assemblies has been maintained identical to that used in Westinghouse fuel assemblies. Compatibility with fuel handling equipment has been established by handling tests and on the basis that the fuel has been loaded into Donald C. Cook Nuclear Plant Unit 2. In addition, essentially identical fuel assemblies having identical upper tie plate configurations have been successfully handled in other plants.

The reactor core consists of a square array of closely packed fuel assemblies. The Donald C. Cook Nuclear Plant Unit 2 reactor has a maximum of 15 assemblies packed side by side and can consist of both ENC and Westinghouse assemblies. During an earthquake the fuel assemblies may interact with each other or with the peripheral core support. A seismic event analysis, therefore, requires that an entire row of assemblies be included to determine the assembly and core baffle interactive loads.

The analyses described in Reference 1, which have been reviewed and approved by the NRC, demonstrate the adequacy of ENC 15x15 fuel assemblies under seismic-LOCA loading. Extensive seismic-LOCA mechanical tests performed with a prototype ENC 17x17 assembly are in Reference 2.

The measured frequency (stiffness) of the two types of assemblies are approximately the same. Therefore, the lateral response of 15x15 or 17x17 fuel assembly arrays to seismic-LOCA excitation should be similar. A comparison of the room temperature strength and stiffness of the two grid spacer types is shown below:

	15x15 Spacer	17x17 Spacer
Strength, lb	2,600	5,180
Stiffness, lb/in	50,000	80,000

Since the 17x17 spacer is substantially stronger than the 15x15 spacer, it would withstand expected seismic-LOCA lateral impacts without significant deformation.

Prototypes of the two assembly types were subjected to the following axial impact loads:

	15x15 Prototype[1]	17x17 Prototype
Maximum impact on lower end, lbs	19,000	50,000
Maximum impact on upper end, lbs	10,000	20,000

In all cases, the test impacts were substantially higher than calculated in Reference 1 for a LOCA event after being corrected for strength at reactor temperature. Deformations measured after testing were insignificant and did not affect control rod withdrawal force[1].

Based upon the above observations, it may be concluded that the ENC 17x17 fuel assembly design can withstand postulated seismic-LOCA forces without affecting safe reactor operation.

To confirm hydraulic compatibility between a Westinghouse and an Exxon Nuclear 17x17 array fuel assemblies, pressure drop tests have been performed in a portable hydraulic test facility. The Westinghouse assembly was tested at the D.C. Cook Nuclear Power Station. The Exxon Nuclear assembly was tested at the Richland site. This test facility is a closed recirculating pressurized water loop designed for operating conditions of 300°F, 200 psig and a 3000 gpm flow rate.

Data were obtained for both test assemblies at coolant temperatures of 185°F, 250°F, and 300°F and in the Reynolds Number range of 50,000 to 275,000. The plenum to plenum pressure drops are compared in Figure 3.5.1-3. The results of this test have shown that Exxon Nuclear's D.C.

Cook 17x17 array fuel is compatible with Westinghouse 17x17 array fuel assemblies. Details are discussed in the Thermal-Hydraulic section of this report.

Guide Tubes

Design Bases

a. Guide tubes shall be of sufficient strength to carry the weight of the fuel assembly, support holddown forces and resist scram forces.

b. The internal diameter of the guide tube shall provide sufficient clearance for control rod insertion to met the required reactor scram control rod insertion time.

c. The guide tubes shall provide for control rod damping in accordance with requirements specified for the Donald C. Cook Nuclear Plant Unit 2 reactor.

d. The guide tube design shall provide for sufficient coolant flow to cool the control rod at any insertion distance. However, the guide tube flow must be restricted to a level which will not significantly affect cooling of the fuel rods.

Design Evaluation

The control rod guide tubes in the fuel assembly provide channels for all types of absorber rods as well as source rods. They are fabricated from a single piece of Zircaloy-4 tubing drawn to two different diameters. The larger diameter section at the top provides a relatively large annular area for rapid Rod Control Cluster (RCC) insertion during a reactor trip and accommodates a small amount of upward coolant flow during normal operation. The bottom of the guide tube is of reduced diameter to produce a dashpot action when the absorber rods approach the end of travel in the guide tubes during a reactor trip. The transition zone at the dashpot section is conical in shape so that there are no rapid changes in diameter in the tube.

The dashpot region is partially plugged at the bottom with a welded end fitting. This end fitting and lower portion of the guide tube fit into a sleeve which passes through and is welded to the bottom spacer and to the lower fitting. The guide tube is mechanically fastened to the lower tie plate via cap screws and the threaded end fittings. Flow holes are provided just above the transition zone to permit the entrance of cooling water during normal operation and to accommodate the outflow of water from the dashpot during a reactor trip. A weep hole is provided in the cap screw.

The structural integrity of the guide tube has been analyzed and laboratory tested. The analytical work has shown that the critical load for buckling on 24 guide tubes, assuming an Euler column, is more than maximum expected column load. In actual practice the guide tubes have some bow, and application of an axial load will increase the amount of bow. Analysis has shown that the increase in bow for maximum predicted loads will not affect integrity, thermal hydraulic conditions, or control rod insertion.[2]

A calculation of control rod motion was performed for the ENC 17x17 design and compatibility with the previously measured rod drop times and design values was evaluated. The forces considered by the analysis method include hydrodynamic drag due to fluid shear and fluid acceleration, mechanical friction, pressure drop due to coolant flow, and gravity including the effects of bouyancy. The calculation is performed with a computer program verified against experimental data taken by ENC. The result of the analysis is as follows:

Parameter	Criteria	Compatibility Information	ENC Design
Time to dashpot (sec)			
Hot, full flow	2.2 max.	1.5	1.4
Hot, zero flow	2.2 max.	1.3	1.2
Velocity at dashpot (ft/s)			
Hot, full flow	-	8.6	9.4
Hot, zero flow	-	10.0	10.3
Total Insertion Time (sec)			
Hot, full flow	-	-	2.3
Hot, zero flow	-	-	2.1
Settling Velocity (ft/s)			
Hot, full flow	0.76 max.	-	0.43
Hot, zero flow	0.76 max.	-	0.49
Force per Absorber Rod (lbf)			
Hot, full flow	150.0	-	37.1
Hot, zero flow	150.0	-	45.0

An analysis of bypass core flow rate through the guide tubes and instrumentation tube shows that the core bypass flow rate is 2.6%.

The mechanical attachment design of the guide tubes and upper tie plate was evaluated from mechanical tests. The tests were performed on hybrided test samples to simulate reactor conditions. Integrity of the joints was maintained to loads greater than 2-1/2 times the assembly weight.

Upper and Lower tie Plates

Design Bases

a. The strength of the tie plates shall be sufficient to withstand the loads resulting from assembly holddown hydraulic forces, and handling and transport forces. The dynamic loads shall be considered equal to 2.5 times static loads[2].

b. The holddown springs shall provide sufficient load to prevent upward motion of the fuel assembly under all normal reactor operating conditions considering the most adverse tolerance conditions.

Design Evaluation

The upper tie plate is a box-like structure which functions as the fuel assembly upper structural element and forms a plenum space where the discharge coolant is mixed and directed toward the flow holes in the upper core plate. The upper tie plate assembly is comprised of a casting, two clamps, four triple leaf springs and four cap screws. All parts, with the exception of the springs and their holddown screws, are constructed of stainless steel. The springs and screws are made from Inconel.

The bottom tie plate 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 tie plate is machined from a stainless steel casting.

The structural integrity of the tie plate has been determined through mechanical tests. Both tie plates required loads in excess of twice the design requirement before signs of yielding occurred[2].

The holddown spring design for the ENC Donald C. Cook Nuclear Plant Unit 2 consists of three leafs, whereas that used on ENC 15x15 has a double leaf. The difference is required to provide greater holddown forces for Donald C. Cook Nuclear Plant Unit 2 assemblies because of the higher pressure drop, while providing increased flexibility.

The weight of an immersed fuel assembly is greater than 1,190 lbs at room temperature and greater than 1,230 lbs at normal reactor operating temperature. The spring constant of the holddown spring system as assembled in the complete tie plate structure is 1,540-1,700 lbs/in cold, determined by test and 1,420-1,570 lbs/in hot, determined through calculation.

Sufficient holddown occurs for the worst case of tolerance stackup during normal operation. At room temperature the minimum spring compression is 0.78 inch: at normal operating temperatures the minimum spring compression is 0.29 inch.

At 20% pump overspeed a net lifting force of 1,800 lbs has been calculated. This load will not cause the holddown springs to yield. The holddown springs are sufficiently far from the active core that no significant in-reactor spring relaxation is expected. The design was confirmed by the fact that there was no evidence of assembly lift during prototype assembly flow tests over a large temperature range at flow rates above design conditions.

Grid Spacers

Design Bases

a. Structural component materials shall retain adequate strength under operating conditions to ensure functional operation throughout the design life of the fuel.

b. The minimum spring force shall be sufficiently large to restrain fuel rod thermal and mechanical bow, to minimize flow induced vibrations, to avoid fatigue failure of the clad tube, and to prevent fretting corrosion at spacer-fuel rod contact points. Irradiation induced stress relaxation shall be considered in establishing the minimum spring force.

c. The maximum spring force shall be less than the contact force at which the calculated contact stresses are equal to the yield strength of the clad tube, or which prevents axial growth of a fuel rod.

Design Evaluation

The Donald C. Cook Nuclear Plant Unit 2 grid spacers consist of interlocking structural Zircaloy-4 grid strips; Inconel spring strips are mechanically secured within the structural strips. Two types of spacers are used; one type without mixing vanes occupies the upper and lower spacer locations; those with mixing vanes occupy the remaining six positions. The uppermost spacer is outside the active fuel zone. The spacers are attached to 24 guide tubes and an instrument tube which pass through them at symmetrical cell locations. Springs and dimples are positioned within each spacer cell such that every fuel rod is in contact with one spring and four support dimples. The Zircaloy-4 structural strips are welded at all intersections and to the enclosing side plates.

Structural characteristics of the grid spacers have been determined both analytically and through mechanical tests. Analysis has shown that the maximum stresses are below design limits for the worst predicted loads. The mechanical tests were of two types: in one test axial loading was applied on the side plates of a grid spacer to assess fuel assembly loading and unloading conditions for irradiated fuel; for the other, a dynamic transverse loading was applied to assess resistance to crushing for seismic evaluation.

The axial loading test showed that spacers, with minimum allowed intersection weld cover, had sufficient strength to meet the design criteria. The dynamic transverse crush test was performed on a spacer loaded with short lengths of tubing. The spacer buckled at a load of 5,500 pounds, equivalent to a 2,900 pound load at temperature, assuming minimum strip thickness at all strips, and repeated loadings. The spacer maintained a coolable geometry throughout the test with no signs of any serious structural damage other than the buckled configuration. The minimum and maximum spring loads measured on assembled spacers have demonstrated that the criteria are maintained throughout the design lifetime of the fuel rods. A minimum load of 2.3 lbs assures that complete dimple lift-off will not occur at EOL. The maximum load of 8.0 pounds does not exceed clad stress limits.

Fuel Rod Design

Design Basis

a. Cladding plastic strain shall not exceed 1.0% from all causes, including fuel swelling, thermal distortion and thermal expansion.

b. Maximum primary membrane stresses resulting from external coolant or internal fission gas pressures shall be less than 2/3 yield or 1/3 ultimate strength based on the volumetric average clad temperature.

c. The cumulative usage factor for cyclic stresses shall not exceed 0.67.

e. The mechanical design of the assembly shall be capable of achieving a maximum assembly average discharge burnup of at least 43,000 MWD/MTU.

Design Evaluation

a. Pellet Configuration

The parabolic radial temperature distribution in a fuel pellet results in a much larger thermal expansion in the center portions of the pellet compared to the edge regions. This condition results in a large differential axial expansion of the pellets which can be accommodated by having dished pellets. ENC's pellet design, as previously described, are dished on both ends which occupy 1% of the pellet volume.

b. Rod Bow Analysis

Special features in the ENC reload design significantly reduce the extent of creep bow. These features include:

1. Thicker cladding

2. Deeper grid spacers which produce a higher rotational restraint on the rods at spacer support points.

3. A 5 point rod support system (4 dimples and 1 spring vs. 4 dimples and 2 springs) which results in less axial restraint.

ENC's rod bow methodology[3,4] has been used to estimate the magnitude of rod bow for ENC 17x17 fuel. The resulting rod bow in terms of fractional rod-to-rod gap closure is given in Figure 3.5.1-4. Gap closures greater than 50% are seen not to occur until assembly exposures above 28,000 MWD/MTU. With ENC's methodology significant rod bow penalty to DNB does not occur until gap closures are in excess of 50%. Thus significant rod bow impacts will not occur until exposures in excess of 28,000 MWD/MTU. Since significant bow penalty increases gradually with exposure beyond 28,000 MWD/MTU and since fissile depletion reduces assembly peaking at high exposure rod bow effects to DNB will not be limiting for ENC 17x17 fuel.

Similarly rod bow impacts to total power peaking will generally be small. At present the 5% total peaking measurement and 3% engineering factor uncertainties for D.C. Cook Unit 2 taken together yield a total bow impact to peaking yields a net uncertainty that is bounded by the 8.15%. Convoluting these uncertainties along with the rod bow impact to peaking yields a net uncertainty that is bounded by the 8.15% uncertainty until high exposure (approximately 34,000 MWD/MTU). Fuel at this exposure will not be limiting so that rod bow will not be limiting with respect to total power peaking.

c. <u>Clad Collapse</u>

In order to guard against the unlikely event that sufficient densification occurs to form pellet column gaps of sufficient size for clad flattening to occur the following evaluation is performed(5).

Creep ovality analysis is performed with the COLAPX code using the existing creep collapse evaluation procedure. Cladding creep down is obtained from the corresponding RODEX2 analysis. The combination of cladding ovality increase and creep down are calculated, and at a rod average burnup of 6,000 MWD/MTU, the combined creep down shall not exceed the initial minimum diametral fuel cladding gap. This will prevent pellet hangups due to cladding creep, allowing the plenum spring to close axial gaps until densification is substantially complete. The calculated value of creep ovality is .00152 in. The calculated value of cladding creepdown is .000608 in. The sum is .00213 in., which is less than the minimum diametral gap of 0.005 in.

d. Cladding Integrity

1. Circumferential Strain

Tests[6,7] on irradiated tubing indicate potential for failure at relatively low mean strains. These tests include tensile, burst and split ring tests, and the data indicate a ductility ranging between 1.2% and 5% at normal reactor operating temperatures. The failures are usually associated with unstable or localized regions of high deformation after some uniform deformation. To prevent cladding failure due to plastic instability and localization of strain, the total mean circumferential cladding strain for steady-state conditions is limited to 1% at end-of-life.

The cladding steady-state strain is evaluated with the RODEX2 code, which is an interactive calculational procedure that considers the thermal-hydraulic environment at the cladding surface, the pressure inside the cladding, and the thermal, mechanical and compositional state of the fuel and cladding. Calculations are performed for the worst expected fuel rod power and fast flux history to determine a conservative history in terms of cladding strain.

In addition to evaluation of the fuel rod steady-state cladding strain, RODEX2 determines the initial conditions for fuel rod power ramping analyses and the fuel rod internal pressures for cladding creep analyses. Pellet density, swelling, densification, and fission gas release models, and cladding and pellet diameters are input to RODEX2 to provide the most conservative subsequent ramping or collapse calculations for the reference fuel rod design.

The fuel rod performance characteristics modeled by the RODEX2 code are:

- Gas release
- Radial Thermal Conduction and Gap Conductance
- Free Rod Volume and Gas Pressure Calculations
- Pellet-cladding Interaction
- Fuel Swelling, Densification, Cracking and Crack Healing
- Cladding Creep Deformation and Irradiation Induced Growth

With the minimum design pellet to cladding gap and the maximum fuel density, the maximum calculated EOL steady-state strain of 0.10% is within the design criteria limit of 1.0 percent.

Volatile fission products combined with high cladding stresses and transient strains is a potential cause of stress corrosion cracking failures. Stress corrosion cracking tests[8,9] have shown that an iodine concentration greater than 10^{-5} to 10^{-6} gm/cm^3 and tensile stresses are both needed to activate the stress corrosion cracking process at cladding inner surface temperatures between 300 and 400°C. At fast fluences below

10^{20} n/cm^2 there is insufficient fission product inventory to allow concentrations that would activate stress corrosion cracking. The strain limit at these conditions is, therefore, set at 1.0% to prevent cladding failure due to plastic instability and localized strain. Power cycling at higher fluences may lead to transient releases of fission products. Where the fission gas composition begins to reach the range of susceptibility to stress corrosion cracking, lower limits on tensile strain are indicated. No power ramp test failures from the Studsvik ramp programs have been observed at a calculated peak circumferential stress level below 70,000 psi. The design limits for transient strains are selected consistent with failure correlations used in the ENC fuel rod performance codes to minimize the potential for stress corrosion cracking failure.

The clad response during ramping power changes was calculated with the RAMPEX code. This code calculates the pellet-cladding interaction during a power ramp. The initial condition are obtained from RODEX2 output. The RAMPEX code considers the thermal condition of the rod in its flow channel and the mechanical interactions that result from fuel creep, crack healing, and cladding creep at any desired axial section in the rod during the power ramp. Maximum hoop stress was determined to be 41,700 psi. The recommended limit is 50,000 psi.

2. Axial Strain

Interference of fuel rods with tie plates can potentially occur as a result of fuel rod growth. Fuel rod length changes with burnup have been measured on irradiated Exxon Nuclear fuel. Based on these fuel data the predicted maximum length change for Donald C. Cook Nuclear Plant Unit 2 fuel rods is 1.35 in. which is 85% of the minimum free space that was designed into the fuel assemblies.

3. Cladding Stress Analyses

The highest rated fuel rods were evaluated to show the relative effects of beginning-of-life (BOL) and end-of-life (EOL) conditions. The stresses produced by primary system pressures, fission gas pressure, thermal gradient, thermal bending, dimple contact forces and cyclic conditions are summarized in ENC's generic design document for 17x17 reload assemblies for Westinghouse plants[2].

4. Strain Fatigue

The number of cumulative strain fatigue cycles is limited to two thirds (2/3) the design strain fatigue life.

Cyclic PCI loading combined with other cyclic loading associated with relatively large changes in power can cause cumulative damage which may eventually lead to fatigue failure. Cyclic loading limits are established to prevent fuel failures due to this mechanism. The design life is based on correlations[10] which give a safety factor of 2 on stress amplitude or a safety factor of 20 on the number of cycles.

The calculations were based upon the duty cycles which conservatively envelope the expected duty cycles of a typical PWR. As for the cladding ramp strain analysis, the power ramp rate was assumed to follow ENC's PREMACCX preconditioning recommendation. Cladding stress amplitudes for the various power cycles were determined from RAMPEX analyses. The initial conditions were obtained from RODEX2 outputs. To account for possible stress concentration in the cladding, an assumed total strain concentration factor of 1.25 was applied to the calculated cyclic cladding stresses. The allowable cycles were determined from the fatigue design curve[10] which considers the effect of maximum mean stress. The total usage factor of 0.26 is less than the design criteria requirement of a maximum cumulative usage factor of 0.67.

e. <u>Fretting Test</u>

The fretting characteristics of fuel rods and spacers were evaluated via a 17x17 proof-of-fabrication assembly flow test at maximum reactor operating conditions for more than 1000 hours. Results of visual inspection are shown in Table 3.5.1-3. The analysis was based on inspection of 12 rods, five of which had had the spacer springs relaxed 100% over the full length of the rod, plus five relaxed to 75% and two rods with nominal spring forces.

<u>Burnable Poison</u>

The burnable poison cluster design consists of the following major components:

1. Holddown assembly made from stainless steel except for the holddown spring which is made from Inconel.

2. Boron rod assemblies consisting of B_4C-Al_2O_3 pellets, Zircaloy-4 cladding and end plugs, and an Inconel plenum spring. The rods are 152.90 inches long overall with a 140. inch long pellet column.

3. Solid plug rods made from stainless steel and are 6.30 inches long.

The burnable poison cluster assemblies can vary in the total number of Al_2O_3-B_4C rods. The cluster can consist of 4, 8, 12, 16, 20, or 24 Al_2O_3-B_4C rodlets, depending on the cycle design and neutronic requirements. The non-Al_2O_3-B_4C rodlet locations contain solid rods as in Item 3 above to maintain proper flow balance for each assembly.

Thermal Design

Design Bases

a. The maximum fuel temperature (at overpower) shall not exceed the fuel melting temperature.

b. The cladding temperature shall be less than:
   850°F Internal Surface
   675°F External Surface
   750°F Volume Average (local)
   based on crud-free surface conditions.

Fuel Temperature Analysis

Fuel temperature for the Exxon Nuclear fuel design are well below the UO_2 incipient fuel melting temperature of 5080 $\pm$ 20°F. The fuel temperature calculation considered the effects of radial neutron flux depression, temperature, density, thermal conductivity, fuel densification and pellet relocation by pellet cracking.

Lyons, et al., (11) thermal conductivity data were used for the UO_2 fuel temperature calculation. The empirical fit of the data used in the analysis is:

$$K(T) = \frac{38.24}{402.4 + T} + 6.126 \times 10^{-13}(T + 273)^3$$

where

$K(T)$ = thermal conductivity, watts/cm-°C, and
T = temperature, °C

Integration of the algebraic thermal conductivity expression between 0 and 2800 degrees centigrade yields 93 watts/cm. This value is recognized to be somewhat less than that derived from other data, which yields an integrated value of 97 watts/cm. The empirical fit of Lyons' UO_2 thermal conductivity data shows a moderate increase in conductivity for fuel temperatures greater than 1500°C which is in basic agreement with in-reactor tests but results in higher predicted fuel temperatures for a given linear heat generation rate.

Lyons' data was obtained for 95% TD fuel. Corrections for the other densities, such as for in-reactor densified fuel, were made using a Loeb type expression:

$$K_p(T) = K(T)_{95}\ [1 - 2.5\ (1 - P)]/0.875$$

where

$K_p(T)$ = Thermal conductivity of fuel with fraction P of theoretical density

$K(T)_{95}$ = thermal conductivity of 95% TD fuel

P = fraction of theoretical density

The film coefficient used to predict the clad wall temperature was based on the Dittus-Boelter(12) correlation for forced convection heat transfer and the Jens-Lottes(13) correlation for heat transfer during subcooled nucleate boiling.

Fuel densification and the corresponding changes in fuel density and pellet radius are calculated from the following empirical expressions:

$$\Delta\rho/\Delta\rho_{max.} = [0.007t] \qquad t < 20$$

$$\Delta\rho/\Delta\rho_{max.} = [0.2198\ \ln(t) - 0.5184] \qquad 20 < t < 1000$$

$$\Delta\rho/\Delta\rho_{max.} = 1.0 \qquad t > 1000$$

where

t = effective full power hours, and

$\Delta\rho_{max.}$ = maximum fuel density change upon completion of densification

and

$$\Delta r = [\Delta\rho/\rho + 2\sigma]\ [r/3]$$

where

$\Delta \rho$ = fuel density change at time t,

σ = standard deviation in the measured probability distribution for pellet density, and

r = nominal as-fabricated pellet radius.

The densification rate expressions were developed from the experimental data of Hanevik, et al.[14]

The model for gap closure which results from pellet cracking and resultant pellet fragment relocation, was based on a detailed investigation of approximately 80 irradiated fuel pellet cross sections which showed that substantial closure of the initial pellet-to-cladding gap occurs after 600 hours of operation or after the first two or three power cycles.

For the Exxon Nuclear Donald C. Cook Nuclear Plant Unit 2 fuel design, the maximum fuel temperature conditions exist at beginning-of-life when the pellet-to-cladding clearance is maximum. The maximum calculated cladding temperature was 642°F at the cladding O.D. and 716°F at the cladding I.D. for 11.66 kw/ft (overpower condition for 3,411 MWt rated power).

Chemical Design

The materials used in the fuel assembly components contained in Batches 6 and 7 fuel are essentially identical to those used in earlier batches. Consequently, the chemical interaction conditions between fuel, cladding, coolant and assembly were not altered during Cycle 4 of Donald C. Cook Nuclear Plant Unit 2, nor were they altered in subsequent cycles.

REFERENCES, SECTION 3.5.1

1. Brown, C.A. "Combined Seismic - LOCA Mechanical Evaluation for Exxon Nuclear 15x15 Reload Fuel for Westinghouse PWRs" Exxon Nuclear Company, XN-76-47, April 1977.

2. Pugh, R.A. "Generic Mechanical Design Report, Exxon 17x17 Fuel Assembly" Exxon Nuclear Company, XN-NF-82-25, Revision 0, April 1982.

3. "Computational Procedure for Evaluating Fuel Rod Bowing", Exxon Nuclear Company, XN-NF-75-32, Supplements 1, 2 and 3, July 1979.

4. Communication Cecil O. Thomas, NRC Division of Licensing to Richard B. Stout, Exxon Nuclear Company, "Acceptance for Referencing of Topical Report", XN-NF-75-32(P), Supplements 1, 2 and 3, February 25, 1983.

5. Ades, M.J., "Qualification of Exxon Nuclear Fuel for Extended Burnup", Exxon Nuclear Company, XN-NF-82-06, March 1982.

6. Bauer, A.A., Lowry, L.M., and Perrin, J.S., "Process on Evaluating Strength and Ductility of Irradiated Zircaloy, Quarterly Progress Report for July through September 1975", BMI-1938, September 1975.

7. Bauer, A.A., Lowry, L.M., Gallaugher, W.J., and Merkworth, A.J., "Evaluating Strength and Ductility of Irradiated Zircaloy - Quarterly Progress Report - January through March 1978, NUREG/CR-0085, BMI-2000, June 1978.

8. Peels, M., Stehle, H., and Steinberg, E., "Out of Pile Testing of Iodine Stress Corrosion Cracking in Zircaloy Tubing in Relation to the PCI-Phenomenon", Fourth International Conference on Zirconium in the Nuclear Industry, 1978.

9. Miller, A.K., Challenger, K.D., Smith, E., Ranjan, G.V., and Cupolla, R.C., "Zircaloy Cladding Deformation and Fracture Analysis", EPRI NP-856, August 1958.

10. O'Donnel, W.J., and Langer, B.F., "Fatigue Design Bases for Zircaloy Components", Nuclear Science and Engineering Volume 20, January 1964.

11. Lyons, M.F. et al., "UO_2 Pellet Thermal Conductivity from Irradiations with Central Melting", GEAP-4624, May 1964.

12. Dittus, F.W. and Boelter, L.M.K., University of California, Berkely, Pub. Trg. 2, 433 (1980).

13. Jens, W.J. and Lottes, P.A., "Analysis of Heat Transfer Burnout, Pressure Drop and Density Data for High Pressure Water", U.S. AEC Report, ANL-4627, (1951).

14. Hanevik, A. et al., "In-Reactor Measurements of Fuel Stack Shortening", Presented by BNES Conference (1973).

TABLE 3.5.1-1

DESCRIPTION OF REGION 6 AND 7 FUEL ASSEMBLIES

Item	Purpose	Material/Rationale
Upper Tie Plate	Provides lifting fixture. Forms plenum space for coolant discharge. Maintains guide tube array.	Cast SS, Grade CF-3 - Strength - Corrosion resistance
Cladding	Contains fission gases and keeps water from contacting fuel.	Zircaloy-4 - Minimize neutron absorption
Fuel Rod End Cap Welds	Provide high quality of fuel rods.	GTAW - Fillet Head - Excellent penetration - Extremely low porosity - High strength integrity
Plenum Spring	Maintain compact fuel column during handling and shipping.	Inconel Wire - Maintain spring load during reactor operation
Plenum Chamber	Collects fission gases. Provides space for axial expansion of fuel.	- Assures that gas pressure will not overstress cladding. - Assures dilution of released fission gases.
Pellet-Cladding Gap	Provide clearance between fuel and cladding.	- Optimized design to maximize fuel rod heat transfer and to minimize pellet-clad interaction from swelling expected at high burnup.

TABLE 3.5.1-1 (continued)

DESCRIPTION OF REGION 6 AND 7 FUEL ASSEMBLIES

Item	Purpose	Material/Rationale
Rod Atmosphere	Heat transfer medium between pellet and clad.	Helium - Good heat transfer characteristics - Provides an easy and reliable leak detection monitoring means
Spacers	Maintain correct rod-to-rod spacing.	Zircaloy-4 Frame, Inconel Springs - Corrosion minimized - Mechanical stability - Spring loads on cladding must be sufficient to minimize lateral and rotational movement of fuel rod but must not cause excessive cladding or spring stress - Spacer must not cause excessive coolant flow resistance
Guide Tubes	Provide channels for control rods, burnable poison rods, source rods.	Zircaloy-4 - Minimize neutron absorption - Minimize differential thermal expansion with fuel rods
Instrumentation Tube	Provide channel for in-core monitoring.	Zircaloy-4 - Minimize neutron absorption - Maintain material continuity with guide tubes
Bottom Tie Plate	Distributes coolant to fuel rods. Maintains guide tube array.	Cast SS, Grade CF-3 - Strength - Corrosion resistance.

TABLE 3.5.1-2

COMPARISON OF MECHANICAL DESIGN VALUES

		Region 6 (ENC)	Regions 3,4,5 (Westinghouse)
A.	FUEL PELLETS		
	Initial Enrichment, w/o U-235	3.65	3.10/3.40/3.40
	Pellet Dish, % of Undished Volume	1.0	NA*
	Average UO_2 Density, % of Theoretical	94	95
	Pellet Diameter, inches	0.3030	0.3225
B.	FUEL ROD		
	Number of Rods Per Assembly	264	264
	Active Length, inches	144.0	144.0
	Rod Pitch, inches	0.496	0.496
	Fill Gas	Helium	Helium
C.	CLADDING		
	Material	Zircaloy-4	Zircaloy-4
	Outside Diameter, inches	0.360	0.374
	Wall Thickness, inches	0.025	0.0225
D.	FUEL ASSEMBLY		
	Geometry	17x17	17x17
	Number of Assemblies	72 ENC	193 Total
	Fuel Assembly Pitch, inches	8.466	8.466
	Envelope at Grid Locations	8.426	8.426

*NA means not available.

TABLE 3.5.1-2 (Continued)

COMPARISON OF MECHANICAL DESIGN VALUES

		Regions 6 & 7 (ENC)	Regions 1 to 5 (Westinghouse)
E.	CONTROL ROD GUIDE TUBE		
	Number/Assembly	24	24
	Material	Zircaloy-4	Zircaloy-4
	ID, inch	0.448	0.450
	OD, inch	0.480	0.482
F.	INSTRUMENTATION TUBE		
	Number/Assembly	1	1
	Material	Zircaloy-4	Zircaloy-4
	ID, inch	0.448	0.450
	OD, inch	0.480	0.482
G.	SPACER GRIDS		
	Number	8	8
	Material	Zircaloy-4/ Inconel	Inconel 718

TABLE 3.5.1-3

FRETTING CORROSION RESULTS

TESTING CONDITIONS

Pressure	2235 psia
Temperature	600°F
Flow	2540 gpm
Coolant	Borated deionized water
Duration	1000 hours

RESULTS

No sign of fretting

Mechanical wear from 0.0 to 0.6 mils

No loosening of fittings or weld failures

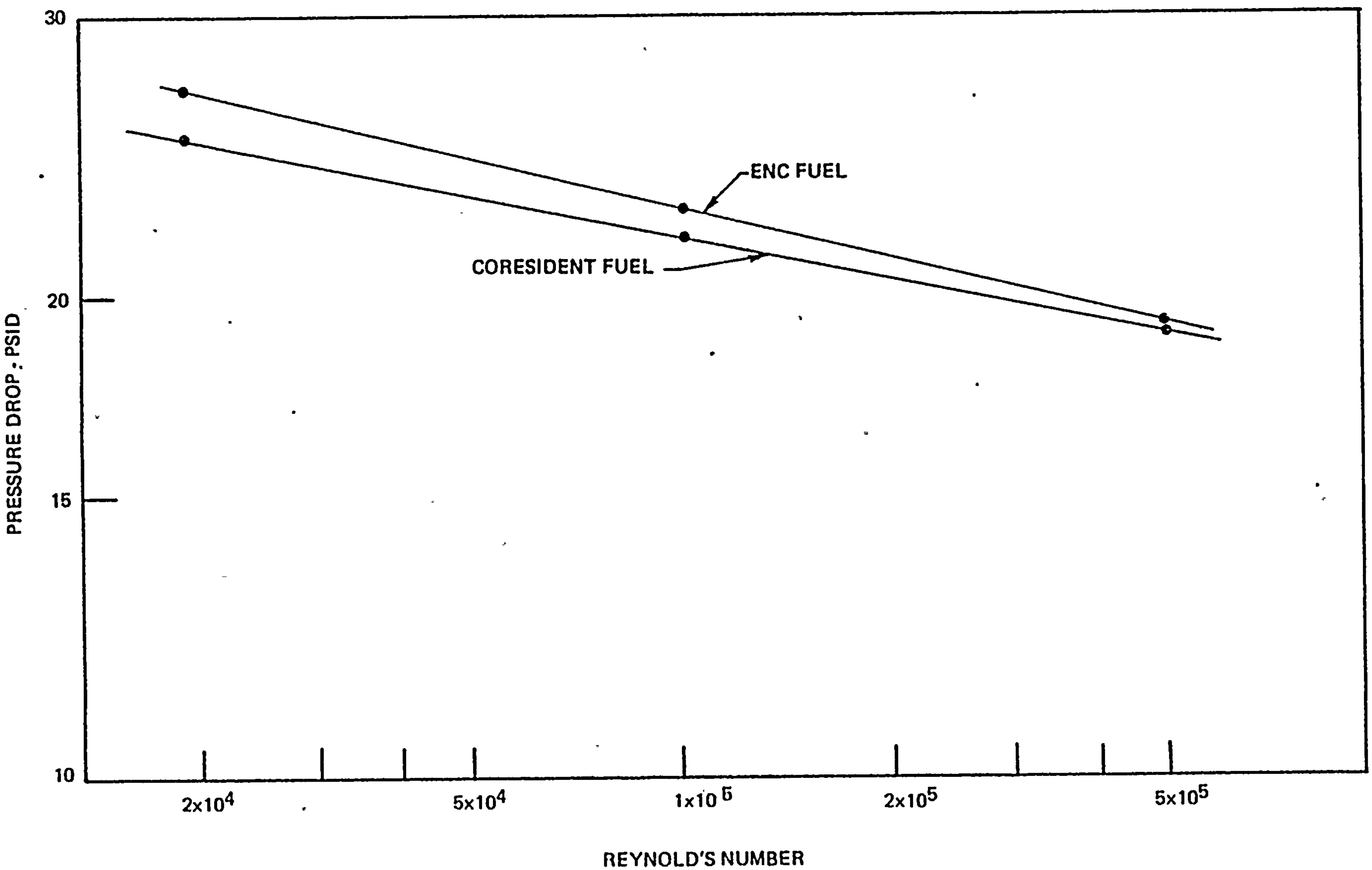


FIGURE 3.5.1-3 BUNDLE OVERALL PRESSURE DROP

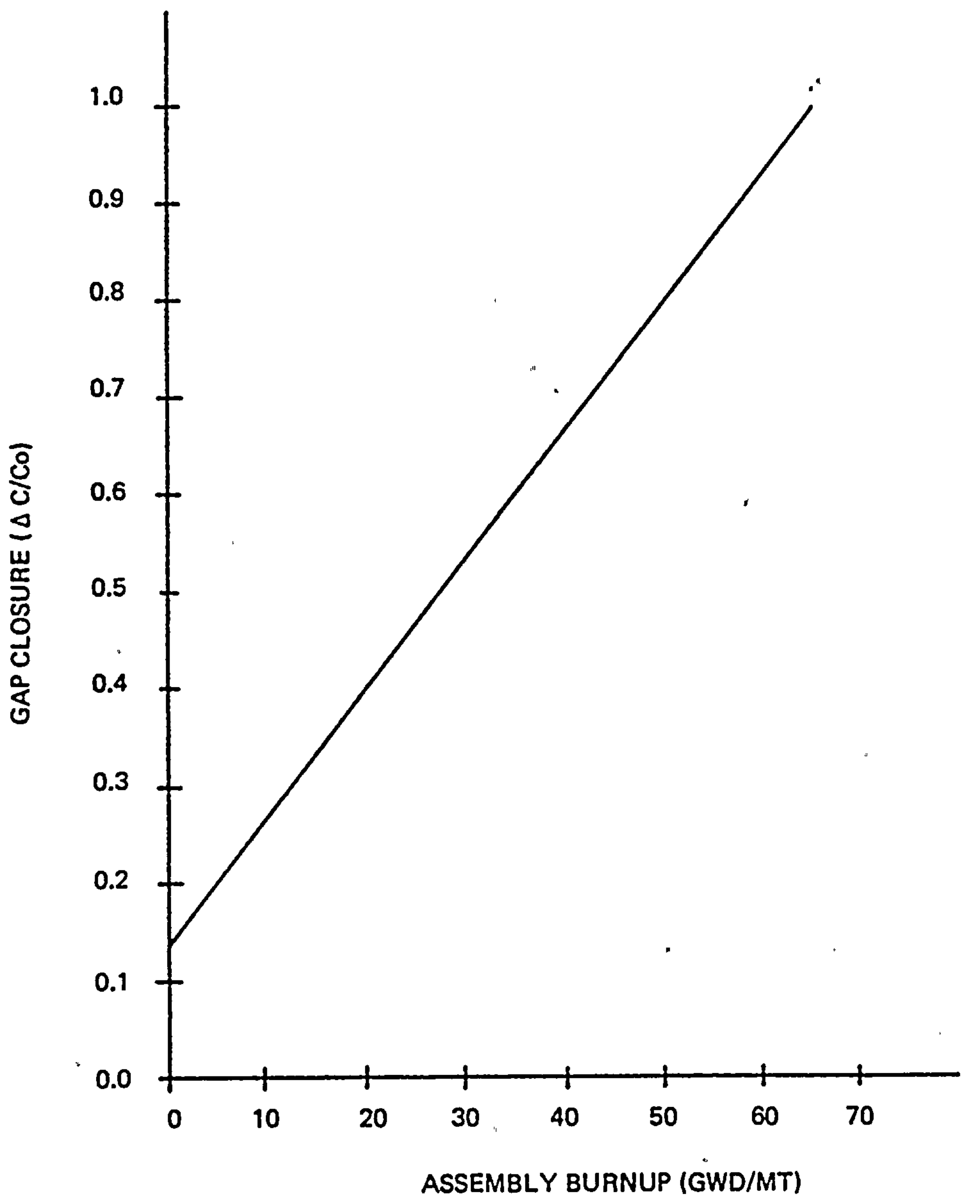


FIGURE 3.5.1-4 ROD BOW FOR ENC 17x17 FUEL

3.5.2 NUCLEAR DESIGN

Exxon Nuclear Company's principal PWR neutronic design tools are the XPOSE code for generating the cross-sections or the basic nuclear parameters, the PDQ7 code for computing reactivity and xy power distributions, and the XTG code for analyses requiring a three-dimensional simulation. The PDQ7 code is a few-group diffusion theory code, which when combined with the HARMONY depletion routine, provides a powerful and flexible core depletion capability. XTG is a simulated two-group diffusion theory code which uses course mesh spacing and can account for important reactivity feedback mechanisms such as power dependent xenon, fuel temperature (Doppler), and moderator temperature. The Exxon Nuclear Company design methods are described briefly in the following subsections and more completely in Reference 1.

Nuclear Data and Computational Methods

Since it would be impractical to provide full descriptions of the computer codes, the reader is referred to the code documents themselves for additional details. See Reference 1.

Nuclear Cross Section Data

Measured neutron cross-sections are the necessary starting point of all neutronic calculations. These are strong functions of neutron energy and exhibit very different values for the various isotopes present in PWR cores.

With a few exceptions the cross-section used by Exxon Nuclear Company are from the national nuclear data file ENDF/B - Version I[2]. The data provides a description of the neutron reaction cross-section over the range from 10 Mev to .0001 ev incident neutron energy. Resonance reactions are described using single level Breit-Wigner resonance parameters. With this exception, the cross-sections are taken to be constant over a small range in energy. The entire energy range from 10 Mev to .0001 ev is described by 345 of the "fine groups."

The Neutron Cross Section Code (XPOSE) And Its Application

Neutron spectra are calculated using the XPOSE code which is an improved version of the LEOPARD[3] code. XPOSE[4] uses the basic nuclear data library to produce spectrum averaged broad group cross-sections over the following energy ranges.

XPOSE BROAD GROUP STRUCTURE

Group No.	Energy Range
1	10 Mev - .821 Mev
2	.821 Mev - 5530 ev
3	5530 ev - 1.855 ev
4	1.855 ev - .0001 ev

The spectrum calculation for energies from .0001 to 1.855 ev is based upon the Wigner-Wilkins approximation as contained in the SOFOCATE[5] code. Spatial thermal self-shielding factors are introduced by means of the Amouyal-Benoist-Horowitz[6] methods where the factors are energy dependent and inherent in the spectral calculation, i.e., they are determined at each energy level. In addition, provision is made to weight non unit-fuel-cell regions such as water channels, control rod guide tubes, and burnable absorber rods by a factor to account for non-uniform thermal neutron flux distributions within the fuel assembly. Two hundred and ninety-five (295) fine groups cover the energy range.

The epithermal slowing down spectrum calculation is performed with 50 MUFT[7] fine energy groups from 10 Mev to 1.855 ev. The resonance cross-sections are Doppler broadened using an input "effective resonance" temperature.

The U-238 resonance absorption is calcuated by a technique which is based upon the experimental measurements of integral absorption by Hellstrand[8].

Treatment of Burnable Absorber

The analytical procedure for representing a burnable absorber rod is ultimately based on the surface boundary conditions for the individual absorber. Specifically, the neutron current-to-flux ratio at the surface of the burnable absorber rod is computed as a function of energy and then averaged over the neutron spectrum as computed by XPOSE for the homogenized fuel assembly. The energy averaged values, thermal and epithermal, of the current-to-flux ratio are then converted into a consistent set of equivalent diffusion theory constants which, when used in diffusion theory calculations, yield the same neutron capture rate as when actual physical constants are used in transport theory calculations.

Since the burnable absorber region homogenized with its clad, the guide tube, and the associated water is represented by a single mesh rectangle in the quarter-core calculations, it is necessary to develop a procedure that will preserve the proper reaction rates as well as worths. The procedure is as follows:

1. Fuel and burnable absorber rod cross-sections are computed by XPOSE. Burnable absorber rod data are for the burnable absorber region only and are generated based on the current-to-flux ratio method described above.

2. A 9 pin array (one burnable absorber pin and eight fuel pins) is described with a 1-D diffusion theory 60 mesh cylinder (40 mesh points describe the burnable absorber cell). Reference fast and thermal reaction rated for the burnable absorber cell are computed in four (three fast and one thermal) broad groups over a range of B-10 concentrations simulating BOL to EOL.

3. A 9 pin array is described with a 3x3 mesh using the mesh description and two group structure to be employed in the quarter-core calculation. The burnable absorber cell macroscopic absorption cross-sections are adjusted so that the fast and thermal reaction rates agree at each B-10 concentration with the explicit calculation in Step 2.

4. Burnable absorber number densities are factored out of the macroscopic cross-sections from Step 3 and micorscopic cross-sections are computed.

5. The resulting microscopic cross-sections for the burnable absorber cell are used in the quarter-core PDQ[9] model and/or in an assembly PDQ model from which macroscopic assembly cross-sections for XTG are calculated.

<u>The Spatial Codes and Their Application</u>

Few-group, spatial calculations are performed primarily with two codes: PDQ7/HARMONY[10] and XTG[11]. Both codes compute an eigenvalue which is the effective multiplication factor and depends on the spectrum average cross-sections throughout the problem space. In general, the diffusion theory codes provide the spatial power distribution and the core reactivity as determined by the cross-sections. XTG includes power-temperature feedback mechanisms through cross-section modification which are currently not available in the PDQ7 models at Exxon Nuclear. These methods are discussed more fully in Reference 1.

Spectrum-averaged few-group cross-sections calculated with XPOSE are utilized in the quarter core PDQ7 model and in an infinite assembly PDQ7 model from which cross-sections for XTG are determined.

PDQ-7/HARMONY

PDQ-7/HARMONY is a few-group diffusion theory code combined with a transmutation subroutine and with a generalized three-dimensional interpolater. This combination makes it possible to handle eigenvalue calculations in one, two or three dimensions and at various burnups. It has some of the most sophisticated numerical routines available included in its structure which gives it a high speed of convergence.

When the depletion routine, HARMONY, is added to PDQ-7, a burnup description of great power and flexibility is available. In particular, it is possible to include exposure dependent variations of microscopic cross-sections of the materials in the assembly. Thus, as the burnup calculation proceeds and the flux spectrum varies, the absorption rate changes, not only because of the spectrum change itself, but because of the accompanying change in the microscopic cross-section for absorption of each of the constituents.

XTG

XTG is a simulated two-group, three-dimensional diffusion theory code utilizing coarse mesh. Diffusion theory is used to solve for the fast group flux in each node. The thermal flux is calculated from the fast flux assuming no thermal leakage to occur between nodes. This permits the use of an iterative solution on the fast group only, which makes it possible to carry out a full three-dimensional core calculation with reasonable computer time usage. The code has a macroscopic burnup model.

UNIT 2, CYCLE 5, NEUTRONIC DESIGN

Analytical Input

The neutronics design methods utilized to calculate the data presented herein are consistent with those described in Reference 1 with primary reliance upon the XTG simulator code.

The burnup history of each of the exposed fuel assemblies was calculated by a three-dimensional, four node per assembly XTG model which was utilized to simulate operation of the core for Cycles 1, 2, 3, and 4. The results of this calculational model are compared for Cycle 4 to a core measured power distribution in Figure 3.5.2-1 and the boron curve in Figure 3.5.2-2.

Calculations for BOC5 utilized the assembly exposures, four values per assembly in 2-D and forty eight (48) values per assembly in 3-D calculated in Cycle 4 at 13,400 MWD/MTU. The 3-D XTG model was verified using the 2-D pin-by-pin PDQ-7/HARMONY model. Axial effects in the 2-D models were accounted for through the buckling term B_z^2.

Design Basis

The nuclear design bases for the Cycle 5 core were as follows:

1. The design shall permit operation within the Technical Specifications for the D. C. Cook Unit 2 nuclear plant.

2. The final Cycle 5 loading pattern shall permit full power (3,411 MWt total power*) operation of the core throughout Cycle 5 reactivity life time of about 17,900 MWD/MT. Power distributions and control

---

* excludes pump heat

rod worth (both shutdown worth and the worth of a potentially ejected rod) are maintained within the ranges analyzed in the Cycle 5 safety analysis.

3. At hot full power (3,411 MWt total power) the peak F_Q^T shall not exceed the limits shown in Figure 3.5.2-8 and the peak $F_{\Delta H}^N$ shall not exceed 1.49 in any single fuel rod through the cycle under nominal operating conditions for Exxon Nuclear Company supplied fuel. For fuel supplied by Westinghouse the allowable F_Q is reduced by 3.4% and the allowable $F_{\Delta H}^N$ by 0.67%.

4. The moderator temperature coefficient is maintained less than or equal to +5 pcm/°F below 70% of rated power and less than or equal to 0 pcm/°F at or above 70% of rated power.

5. The scram worth of all rods minus the most reactive rod shall exceed BOC and EOC shutdown requirements.

Nuclear Design Description

The reactor core consists of 193 assemblies, each having 17x17 fuel rod array. Each assembly contains 264 fuel rods, 24 RCC guide tubes, and 1 instrumentation tube. The fuel rods consist of slightly enriched UO_2 pellets inserted into zircaloy tubes. The RCC guide tubes and the instrumentation tube are also made of zircaloy. Each ENC assembly contains eight zircaloy spacers with Inconel springs; seven of the spacers are located within the active fuel region.

The Cycle 5 loading pattern is shown in Figure 3.3.2-3 with assemblies identified by Fabrication ID and the burnable absorber configuration. The core consists of 92 fresh ENC assemblies with an average enrichment of 3.64 w/o U-235, 29 exposed Westinghouse assemblies and 72 exposed ENC assemblies. A low radial leakage fuel management plan has been developed and results in the scatter-loading of the fresh fuel throughout the core with the fresh assemblies loaded in the core interior containing Al_2O_3-B_4C burnable absorber rods. The exposed fuel is also scatter-loaded in the center in a manner to control the power peaking. The Al_2O_3-B_4C burnable absorber rods contain 0.026 gm/in of B-10 and 1040 of these rods are distributed among 72 fresh assemblies loaded in the core interior. Pertinent fuel assembly parameters for the Cycle 5 fuel are depicted in Table 3.5.2-1.

Physics Characteristics

The neutronics characteristics of the Cycle 5 core are compared with those of Cycle 4 and are presented in Table 3.5.2-2. The data presented in the table indicates the neutronic similarity between Cycles 4 and 5. The reactivity coefficients of the Cycle 5 core are bounded by the coefficients used in the safety analysis.

The boron letdown curve for Cycle 5 is shown in Figure 3.5.2-4. The BOC5 xenon free critical boron concentration is calculated to be 1491 ppm. At 100 MWD/MT, equilibrium xenon, the critical boron concentration is 1149 ppm. The Cycle 5 length is projected to be 17,900 MWD/MT $\pm$ 300 MWD/MT at a core power of 3,411 MWt with 10 ppm soluble boron remaining.

Power Distribution Considerations

Representative calculated power maps for Cycle 5 are shown in Figures 3.5.2-5 and 3.5.2-6 for BOC and EOC conditions, respectively. The radial power distributions are representative of the all rods out, equilibrium xenon configurations. A comparison between predicted power distribution (XTG) and measured power distribution (Flux Map 205-60) is shown in Figure 3.5.2-7. The hot full power allowable F_Q as a function of core height including the axial dependent K(z) penalty is shown in Figure 3.5.2-8 for Exxon Nuclear Company supplied fuel.

Control Rod Reactivity Requirements

Detailed calculations of shutdown margins for Cycle 5 are compared with Cycle 4 data in Table 3.5.2-3. The D. C. Cook Unit 2 nuclear plant Technical Specifications require a minimum required shutdown margin of 1,600 pcm at BOC and EOC. The Cycle 5 analysis indicates excess shutdown margin of 1008 pcm at BOC and 721 at the EOC. The Cycle 4 analysis indicated an excess shutdown margin of 722 pcm at BOC and 734 pcm at EOC.

The control rod groups and insertion limits for Cycle 5 will remain unchanged from Cycle 4. With these limits the nominal worth of the controlbank, D-Bank, inserted to the insertion limits at HFP is 149 pcm at BOC and 272 pcm at EOC. The control rod shutdown requirements allow for a HFP D-Bank insertion equivalent to 400 pcm and 500 pcm at BOC and EOC, respectively.

Moderator Temperature Coefficient

The Technical Specifications require that the moderator temperature coefficient be less than or equal to +5 pcm/°F below 70% of rated power and less than or equal to 0 pcm/°F at or above 70% power. The HZP, ARO moderator temperature coefficient is calculated to be +3.0±2 pcm/°F and meets the Technical Specification limit below 70% power. The moderator temperature coefficient at or above 70% power is calculated to be less than 0 pcm/°F and also meets the Technical Specification requirements.

REFERENCES, SECTION 3.5.2

1. "Exxon Nuclear Neutronics Design Methods for Pressurized Water Reactors", XN-75-27, and Supplements 1 and 2, Exxon Nuclear Company, December 1977.

2. H. C. Honeck, "ENDF/B - Specifications for An Evaluated Nuclear Data File for Reactor Applications," BNL-5006, (May, 1966).

3. R. F. Barry, "LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code," WCAP-3269-29, Westinghouse Electric Corporation (1963).

4. F. B. Skogen, "XPOSE - The Exxon Nuclear Revised LEOPARD," XN-CC-21 (Revision 2), Exxon Nuclear Company, (April, 1975).

5. H. Amster and R. Suarez, "The Calculation of Thermal Constants Averaged over a Wigner-Wilkins Flux Spectrum: Description of the SOFOCATE Code," WAPD-TM-39 (January, 1957).

6. A. Amouyal, P. Benoist, J. Horowitz, "New Method of Determining the Thermal Utilization Factor in a Unit Cell," J. Nuclear Energy, Vol. 6 (1957).

7. H. Bohl, E. Gelbard, and G. Ryan "MUFT-4 - Fast Neutron Spectrum Code for the IBM-704," WAPD-TM-72 (July,1957).

8. E. Hellstrand, P. Blomberg and S. Horner, Nucl. Sci. Eng., 8 497 (1960).

9. W. R. Cadwell, "PDQ-7 Reference Manual," WAPD-TM-678, Westinghouse Electric Corporation, (January, 1967).

10. R. J. Breen, O. J. Marlowe, C. J. Pfeifer, "HARMONY: System for Nuclear Reactor Depletion Computation, "WAPD-TM-478, Westinghouse Electric Corporation, (January, 1965).

11. R. B. Stout, "XTG - A Two-Group Three-Dimensional Reactor Simulator Utilizing Coarse Mesh Spacing," XN-CC-28 (Revision 5), Exxon Nuclear Company, (July, 1979).

Table 3.5.2-1

D. C. Cook Unit 2, Principal Characteristics for Nuclear Analysis of Cycle 5 Fuel

	Region 5	Region 6	Region 7
Nominal Enrichment (w/o)	3.40	3.65	3.64
Nominal Density (% TD)	95	94	94
Pellet OD (in)	.3225	.3030	.3030
Clad OD (in)	.374	.360	.360
Diametral Gap (in)	.0065	.0070	.0070
Clad Thickness (in)	.0225	.0250	.0250
Rod Pitch (in)	.496	.496	.496
Spacer Material	Inconel	Bi-Metallic	Bi-Metallic
Fuel Supplier	W	ENC	ENC
Fuel Stack Height Nominal (in)	144	144	144
Number of Assemblies	29	72	92
Regionwise Loading (MTU)	13.286	29.077	37.154
Exposure (MWD/MT)			
BOC5	24,069	16,368	0
EOC5	34,866	35,410	19,546
Incremental	10,797	19,042	19,546

Table 3.5.2-2

D. C. Cook Unit 2, Neutronics Characteristics of Cycle 5 Compared With Cycle 4 Data

	Cycle 4		Cycle 5	
	BOC	EOC	BOC	EOC
Critical Boron				
HFP, ARO, Eq. Xenon (ppm)	989(b)	10(b)	1,149	10
HZP, ARO, No Xenon (ppm)	1,465(a)	--------	1,569	--------
Moderator Temperature Coefficient				
HFP, (pcm/°F)	-4.0 (b)	-27.5(b)	-2.1	-26.3
HZP, (pcm/°F)	-0.97(a)	-21.9(b)	+3.0	-21.1
Isothermal Temperature Coefficient				
HFP, (pcm/°F)	-5.4 (b)	-29.2(b)	-3.4	-27.8
HZP, (pcm/°F)	-2.86(a)	-23.6(b)	+1.3	-23.0
Doppler Coefficient (pcm/°F)	-1.4	-1.6	-1.3	-1.5
Boron Worth, (pcm/ppm)				
HFP	-7.7 (b)	-8.7 (b)	-8.0	-9.6
HZP	-8.95(a)	-10.9(b)	-9.4	-11.7
Total Nuclear Peaking Factor				
F^N_Q, HFP, Equilibrium Xenon	1.59 (a)	1.55 (b)	1.64	1.54
Delayed Neutron Fraction	.0057	.0051	.0062	.0051
Control Rod Worth of All Rods In Minus Most Reactive Rod, HZP, (pcm)	5,525	6,093	6,301	6,172
Excess Shutdown Margin, (pcm)(c)	722	734	1,008	721

(a) Measured data

(b) ENC calculated

(c) Shutdown margin evaluation based on the most adverse combination of power level and rod insertion

Table 3.5.2-3

D. C. Cook Unit 2 Control Rod Shutdown Margins and Requirements of Cycle 5 Compared to Cycle 4

	Cycle 4		Cycle 5	
	BOC	EOC	BOC	EOC
Control Rod Worth (HZP), pcm				
All Rods Inserted (ARI)	6,348	6,888	7,065	7,279
ARI Less Most Reactive (N-1)	5,525	6,093	6,065	6,079
N-1 Less 10% Allowance [(N-1)*.9)]	4,972	5,484	5,458	5,471
Reactivity Insertion, pcm[(a)]				
Power Defect (Moderator+Doppler)	400	500	400	500
Flux Redistribution	600	600	600	600
Void	50	50	50	50
Sum of the Above Three	1,050	1,150	1,050	1,150
Rod Insertion Allowance	1,600	2,000	1,800	2,000
Total Requirements	2,650	3,150	2,850	3,150
Shutdown Margin (N-1)*.9 - Total Requirements	2,322	2,334	2,608	2,321
Required Shutdown Margin	1600[(b)]	1600[(b)]	1600[(b)]	1600[(b)]
Excess Shutdown Margin	722	734	1,008	721

(a) The reactivity insertion allowance assumes the most adverse combination of power level and rod insertion. The BOC shutdown margin is increased at HFP conditions and the EOC shutdowm margin remains unaffected at HFP conditions.

(b) Technical Specification limit.

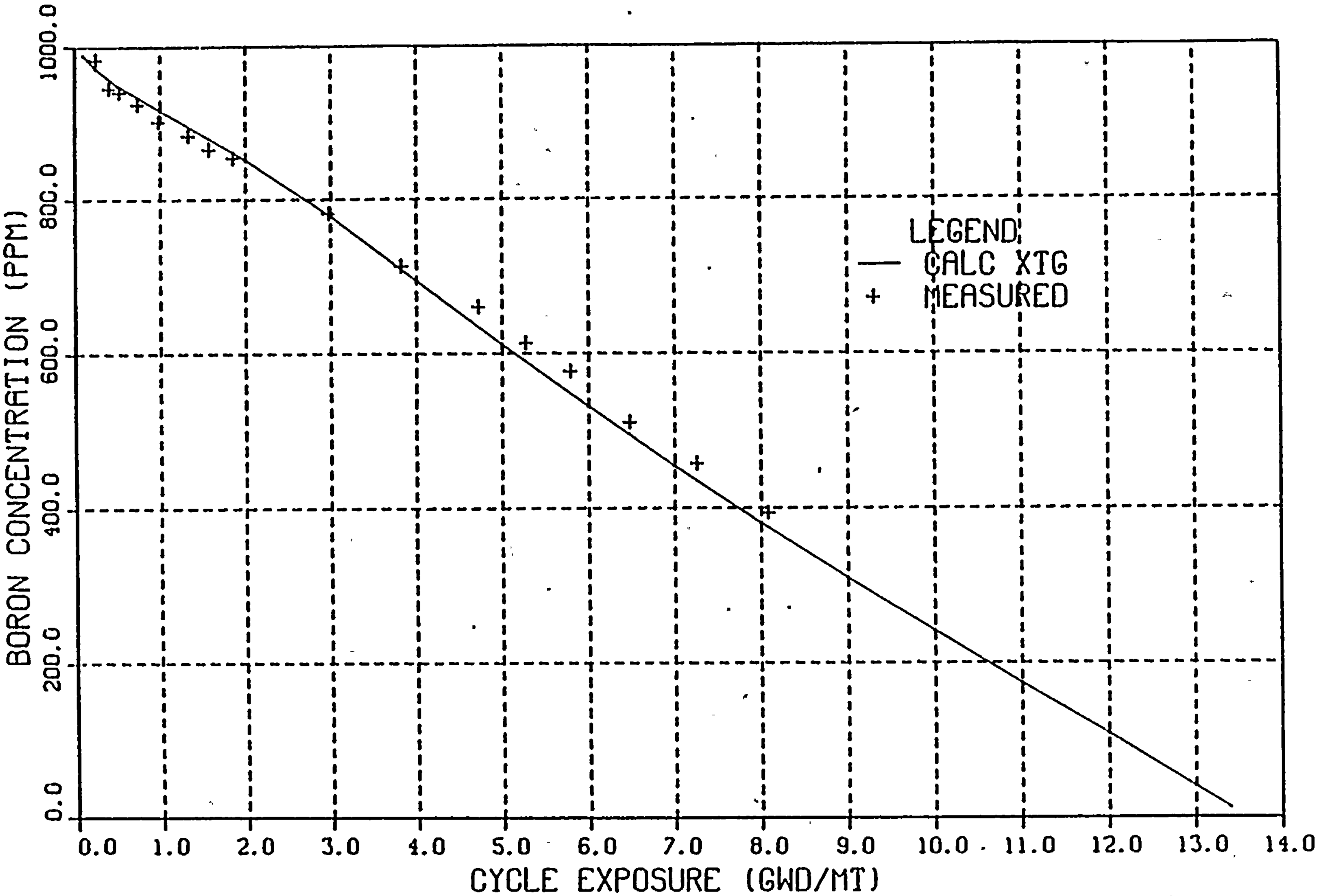
	H	G	F	E	D	C	B	A
8	.856 .848 +0.9	.985 .966 +2.0	.975 .964 +1.1	1.046 1.042 +0.4	.971 .983 -1.2	1.069 1.086 -1.6	1.008 1.019 -1.1	.901 .859 +4.9
9	.982 .968 +1.4	1.079 1.064 +1.4	1.218 1.186 +2.7	1.071 1.069 +0.2	1.218 1.206 +1.0	.997 1.002 -0.5	1.128 1.125 +0.3	.742 .737 +0.7
10	.974 .972 +0.2	1.220 1.187 +2.8	1.081 1.079 +0.2	1.073 1.080 -0.6	1.104 1.123 -1.7	1.234 1.243 -0.7	1.020 1.039 -1.8	.862 .835 +3.2
11	1.049 1.041 +0.8	1.076 1.065 +1.0	1.095 1.072 +2.1	1.093 1.099 -0.5	1.246 1.249 -0.2	1.024 1.058 -3.2	1.107 1.126 -1.7	.554 .564 -1.8
12	.970 .983 -1.3	1.219 1.196 +1.9	1.105 1.118 -1.2	1.246 1.240 +0.5	.990 1.030 -3.9	1.175 1.195 -1.7	.758 .766 -1.0	
13	1.068 1.059 +0.9	.980 .986 -0.6	1.227 1.221 +0.5	1.023 1.051 -2.7	1.173 1.190 -1.4	1.019 1.031 -1.2	.396 .401 -1.2	
14	1.007 1.010 -0.3	1.124 1.109 +1.4	.999 1.030 -3.0	1.102 1.105 -0.3	.755 .757 -0.3	.395 .395 0.0		
15	.903 .852 +6.0	.748 .735 +1.8	.857 .822 +4.3	.551 .558 -1.3				

Calculated (XTGPWR)
Measured Assembly Power
$\frac{C-M}{M} \times 100$

	Calculated	Measured	% Diff.
$F^N_{\Delta H}$	1.354	1.343	+0.8
F^N_Q	1.565	1.557	+0.5

Figure 3.5.2-1

D. C. Cook Unit 2, Cycle 4, Power Distribution Comparison to Map 204-46, 100% Power, Bank D @220 Steps, 7,752 MWD/MT



3.5-49

Figure 3.5.2-2

D. C. Cook Unit 2, Cycle 4 Boron Letdown Curve

	R	P	N	M	L	K	J	H	G	F	E	D	C	B	A
1					M2 R47	+	+	+	+	+	D2 R8				
2			J7 R92	F15 S03	a	D13 S27	c	J1 R73	c	M13 S13	a	K15 S01	H3 R46		
3		N8 R19	a	b	L2 S21	c	L4 S51	H1 S06	E4 S46	c	E2 S57	b	a	G7 R70	
4		A10 S10	b	P7 S45	d	K3 S66	c	J12 S37	c	F3 S32	d	B7 S23	b	R10 S08	
5	P4 R78	a	P5 S63	d	J6 S48	c	E15 R6	c	L15 R65	c	G6 S41	d	B5 S61	a	B4 R23
6	+	C12 S19	c	N6 S30	c	K7 S31	d	G4 S52	d	F7 S35	c	C6 S53	c	N12 S17	+
7	+	c	M5 S28	c	A11 R37	d	J2 S39	N13 S54	G2 S42	d	R11 R89	c	D5 S43	c	+
8	+	R7 R4	R8 S07	D7 S72	c	M9 S34	C13 S58	J15 R54	N3 S20	D9 S15	c	M7 S49	A8 S05	A9 R9	+
9	+	c	M11 S65	c	A5 R2	d	J14 S68	C3 S56	G14 S69	d	R5 R57	c	D11 S70	c	+
10	+	C4 S12	c	N10 S25	c	K9 S64	d	G12 S22	d	F9 S38	c	C10 S33	c	N4 S50	+
11	P12 R60	a	P11 S60	d	J10 S47	c	E1 R33	c	L1 R81	c	G10 S26	d	B11 S62	a	B12 R62
12		A6 S02	b	P9 S29	d	K13 S36	c	J4 S71	c	F13 S44	d	B9 S40	b	R6 S09	
13		J9 R1	a	b	L14 S59	c	L12 S67	H15 S24	E12 S14	c	E14 S55	b	a	C8 R52	
14			H13 R42	F1 S11	a	D3 S18	c	G15 R36	c	M3 S16	a	K1 S04	G9 R71		
15					M14 R49	+	+	+	+	+	D14 R3				

Previous Core Location
Fabrication ID

+ Fresh Fuel Assembly, No BA Pins
a Fresh Fuel Assembly, 4 BA Pins
b Fresh Fuel Assembly, 12 BA Pins
c Fresh Fuel Assembly, 16 BA Pins
d Fresh Fuel Assembly, 20 BA Pins

Figure 3.5.2-3 D. C. Cook Unit 2, Cycle 5, Full Core Loading Pattern

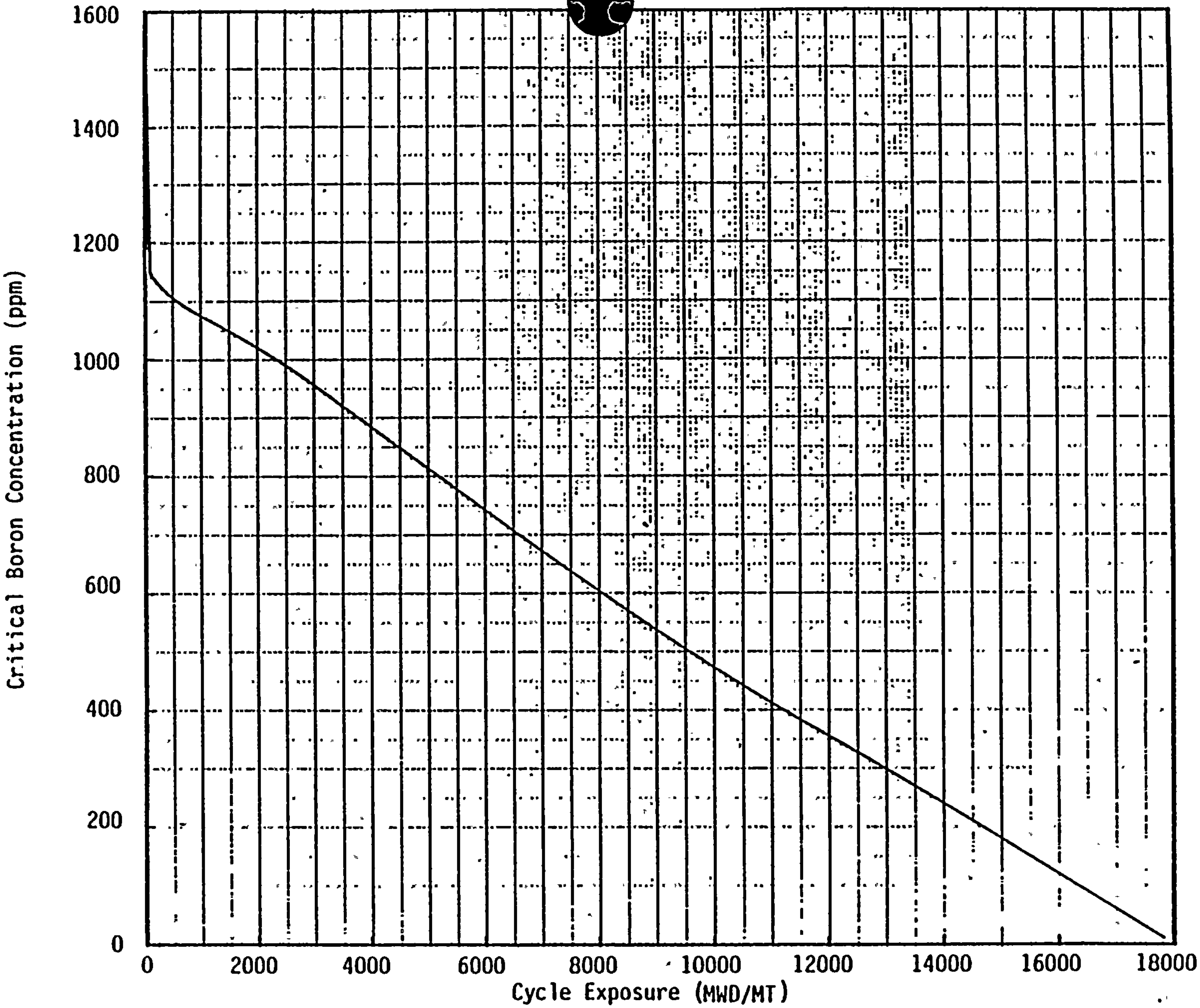


Figure 3.5.2-4 D. C. Cook Unit 2, Cycle 5, Boron Letdown Curve

	H	G	F	E	D	C	B	A
8	1.049	1.169	1.119	1.159	1.136	1.177	.955	.993
9	1.206	1.164	1.109	1.042	1.175	1.112	1.098	.982
10	1.116	1.110	1.099	1.154	1.102	1.148	1.021	.856
11	1.160	1.044	1.156	1.090	1.084	1.072	1.054	.409
12	1.141	1.177	1.106	1.085	1.071	1.050	.681	
13	1.177	1.113	1.151	1.075	1.050	.886	.316	
14	.954	1.099	1.023	1.056	.682	.309		
15	.993	.982	.857	.410				

Assembly Relative Power

Peak Assembly = 1.206 (H9)

Pin $F^N_{\Delta H}$ = 1.323 (H9)

Peak F^N_Q = 1.644 (G15)

Figure 3.5.2-5 D.C. Cook Unit 2, Cycle 5, Relative Power Distribution, 100 MWD/MT, 1149 ppm, 3411 MWt, ARO

	H	G	F	E	D	C	B	A
8	.904	1.002	1.075	1.233	1.072	1.035	.894	.841
9	1.025	1.054	1.216	1.094	1.221	1.036	1.077	.850
10	1.073	1.217	1.119	1.259	1.108	1.190	.954	.777
11	1.233	1.095	1.260	1.127	1.235	1.057	.997	.434
12	1.075	1.222	1.110	1.235	1.100	1.122	.732	
13	1.035	1.036	1.190	1.058	1.121	.955	.396	
14	.893	1.077	.954	.997	.732	.387		
15	.841	.850	.777	.434				

Assembly Relative Power

Peak Assembly = 1.260 (F11)

Pin $F^N_{\Delta H}$ = 1.369 (F11)

Peak F^N_Q = 1.536 (F11)

Figure 3.5.2-6 D. C. Cook Unit 2, Cycle 5, Relative Power Distribution
17,900 MWD/MT, 10 ppm, 3411 MWD/MT, ARO

	H	G	F	E	D	C	B	A
8	0.921 0.932 1.2	1.034 1.043 0.9	1.112 1.100 -1.1	1.224 1.224 0.0	1.109 1.100 -0.8	1.046 1.073 2.6	0.893 0.893 0.0	0.810 0.861 6.3
9	1.033 1.043 1.0	1.090 1.090 0.0	1.188 1.194 0.5	1.122 1.093 -2.6	1.204 1.214 0.8	1.055 1.051 -0.4	1.048 1.056 0.8	0.819 0.865 5.6
10	1.118 1.096 -2.0	1.190 1.194 0.3	1.177 1.143 -2.9	1.259 1.251 -0.6	1.141 1.117 -2.1	1.167 1.171 0.3	0.972 0.956 -1.6	0.737 0.784 6.4
11	1.224 1.224 0.0	1.124 1.094 -2.7	1.260 1.251 -0.7	1.178 1.146 -2.7	1.214 1.196 -1.5	1.081 1.060 -1.9	1.005 1.004 -0.1	0.415 0.420 1.2
12	1.117 1.104 -1.2	1.210 1.215 0.4	1.145 1.119 -2.3	1.214 1.197 -1.4	1.136 1.102 -3.0	1.117 1.102 -1.3	0.706 0.719 1.8	
13	1.054 1.071 1.6	1.056 1.051 -0.5	1.166 1.172 0.5	1.085 1.062 -2.1	1.123 1.104 -1.7	0.921 0.944 2.5	0.368 0.373 1.4	
14	0.893 0.892 -0.1	1.047 1.056 0.9	0.973 0.957 -1.6	1.007 1.006 -0.1	0.714 0.726 1.7	0.383 0.396 3.4		
15	0.809 0.861 6.4	0.820 0.865 5.5	0.737 0.785 6.5	0.415 0.420 1.2				

Measured Assembly Power
Calculated (XTGPWR)
$\left(\frac{C-M}{M}\right) \times 100$

	Calculated	Measured	% Diff.
$F^N_{\Delta H}$	1.359	1.316	+3.3
F^N_Q	1.560	1.525	+2.3

Figure 3.5.2-7 D. C. Cook Unit 2, Cycle 5, Power Distribution Comparison to Map 205-60, 100% Power, Bank D @ 221 Steps, 11,454 MWd/MTU

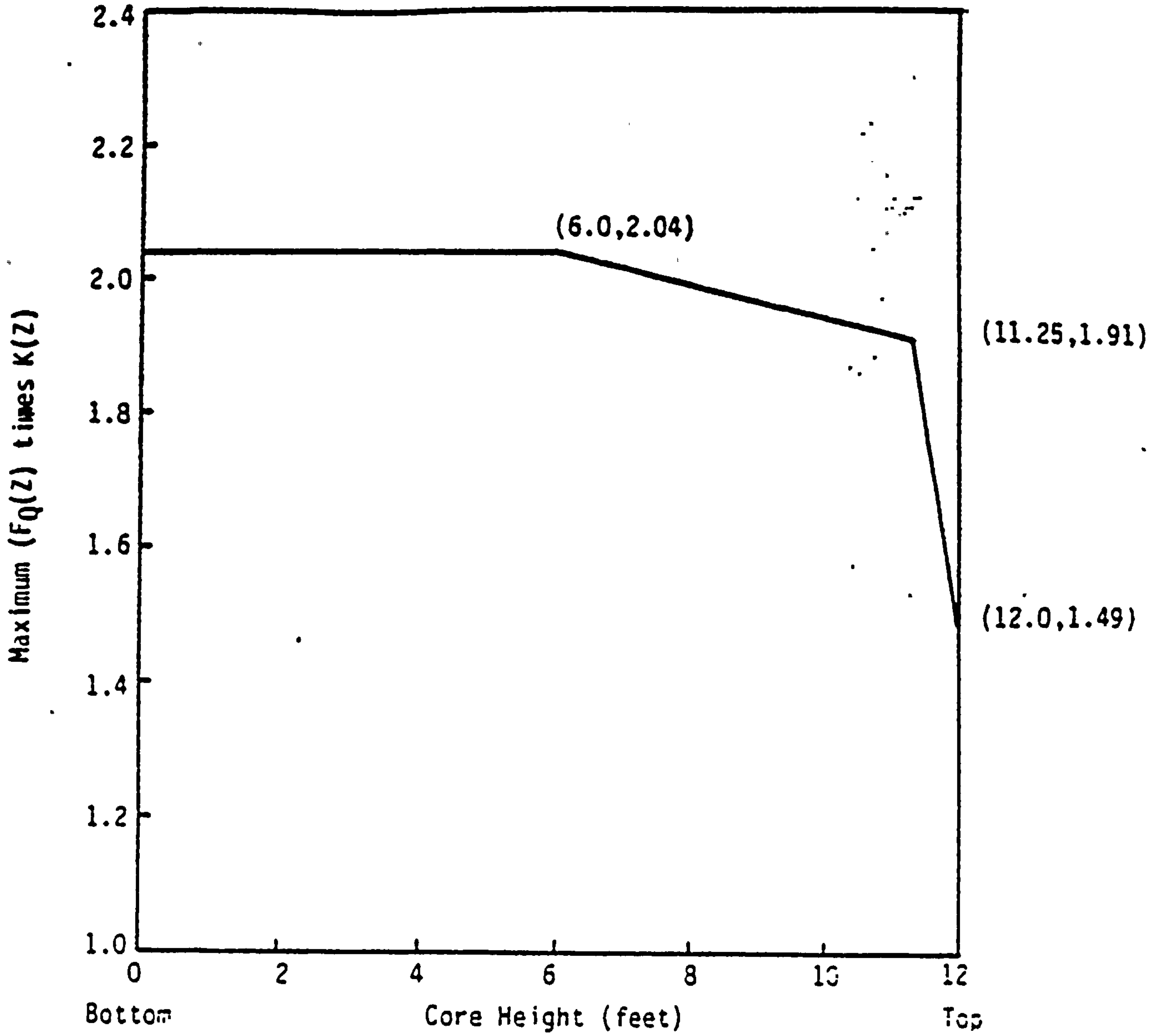


Figure 3.5.2-8 Maximum (F_Q Times K(Z)) versus Axial Height for Exxon Nuclear Fuel for Cycle 4 Operation

3.5.3 THERMAL-HYDRAULIC DESIGN

Design Bases

The thermal hydraulic design performance requirements for the Exxon Nuclear 17x17 design were as follows:

1. The minimum departure from nucleate boiling ratio (MDNBR) is > 1.17 as calculated using the XNB correlation.[1] In addition, an adjustment of 2% on the MDNBR is included for mixed cores containing hydraulically different fuel assemblies.
2. The fuel is thermally and hydraulically compatible with the Westinghouse fuel in the core at the time of insertion of the reload fuel.

Design Analysis

The predicted steady-state thermal-hydraulic performance of the ENC Donald C. Cook Nuclear Plant Unit 2 reload fuel design satisfied all of the design basis requirements. The thermal-hydraulic analysis was performed at 118.5% of rated power (3,411 MWt). The analysis in this section was performed with a total power peaking factor (F_Q^T) of 2.58 which includes a 1.03 engineering factor and a 1.01 densification factor corresponding to an allowed factor of 2.48. The Donald C. Cook Nuclear Plant Unit 2 was licensed[2] for operation in Cycles 4 and 5 with a maximum peaking factor (F_Q) of 2.04 for ENC reload fuel and 1.97 for Westinghouse fuel. The larger nuclear heat flux factor was used in the analysis to demonstrate that adequate thermal margins exist at the increased peaking in order to anticipate future changes in allowable peaking. The results of the analysis are summarized in Table 3.5.3-1.

Minimum Departure From Nucleate Boiling Ratio (MDNBR)

The MDNBR of the Donald C. Cook Nuclear Plant Unit 2 fuel at 118.5% of rated power was calculated to be 1.68 for the Westinghouse fuel and 1.42 for the ENC reload fuel at the design total peaking factor of 2.58 for both fuel types. The smaller rod size and larger flow area of ENC fuel account for a large part of the reduced MDNBR relative to Westinghouse fuel. The MDNBR calculation was performed with standard calculational techniques described in Reference 3. The MDNBR calcualtion was based on the XNB critical heat flux correlation with correction factors for non-uniform axial heat flux profile and unheated subchannel boundaries.

Thermal Hydraulic Compatibility

The hydraulic compatibility tests of Westinghouse and ENC 17x17 fuel, are described in Section 3.5.1. The hydraulic characteristics of the ENC 17x 17 fuel are calculated to closely match those of the Westinghouse fuel. At a flow rate corresponding to nominal reactor operating conditions, the plenum-to-plenum pressure drop of the ENC fuel assembly is approximately equal to the Westinghouse fuel assembly. Between the tie plates, the Westinghouse fuel has about 1.2 psia less pressure drop than the ENC fuel assembly. This small difference in assembly pressure drop has negligible effect upon the margin to DNB.

REFERENCES, SECTION 3.5.3

1. Macduff, R. B., "Exxon Nuclear DNB Correlation for PWR Fuel Designs", XN-NF-621(P), Rev. 1, April 1982.

2. "D.C. Cook Unit 2, Technical Specifications, License No. DPR-74, Appendix A'.

3. Lindquist, T. R., et al., "Application of Exxon Nuclear Company PWR Thermal Margin Methodology to Mixed Core Configurations", XN-NF-82-21(P), Rev. 1., September 1982.

TABLE 3.5.3-1

THERMAL HYDRAULIC DESIGN VALUES USED IN EVALUATION

Rated Heat Output	3,411 MWt
Maximum Overpower	18.5%
Heat Generated in Fuel	97.4%
Nominal Design Pressure	2,250 psi
Design Inlet Temperature	543.1 ^{o}F
Average Core Temperature	574.1 ^{o}F
Total Reactor Coolant Flow	142.7×10^6 lbs/hr
Active Coolant Flow	136.3×10^6 lbs/hr
Average Mass Velocity(1)	2.525×10^6 lbs/hr-ft^2
Average Coolant Velocity Along Fuel Rods(1)	15.5 ft/sec
Active Heat Transfer Surface Area(1)	57,625 ft^2
Average Heat Flux	197,562 Btu/hr-ft^2
Maximum Heat Flux(1,3)	601,446 Btu/hr-ft^2
Maximum LHGR(3)	16.61 kw/ft
Average LHGR	5.456 kw/ft
MDNBR - ENC(2,3)	1.42
MDNBR - Westinghouse(2,3)	1.68

(1)Core is fueled with all ENC assemblies.

(2)Mixed core with both ENC and Westinghouse fuel.

(3)At overpower.

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), ft3 (estimated)	12,500
System liquid volume, including pressurizer and surge line (ambient conditions), ft3	11,892
System liquid volume, including pressurizer max. guaranteed power, ft3 (estimated)	11,780
Total Reactor heat output (100% power) Btu/hr	11,089 x 106 (Unit 1) (3250 MWt) 11,641 x 106 (Unit 2) (3411 MWt)

	Unit 1	Unit 2
Reactor vessel coolant temperature at full power:		
Inlet, nominal, oF	536.3	541.27
Outlet,, oF	599.3	606.35
Coolant temperature rise in vessel at full power, avg., oF	63.0	64.8
Total coolant flow rate, lb/hr	135.6 x 106	134.6 x 106
(*) Steam pressure at full power, psia	758	820
Steam Temp. @ full power, oF	512.1	521.1
Total Reactor Coolant Volume at ambient conditions, ft3	12,438	12,438

(*)

Equivalent to 88,500 gpm/loop. This value is the one used in non-ITDP transients. The value used in the analysis of ITDP transients is 142.7 x 106 lb/hr. Measured values are typically 146.0 x 106 lb/hr.

July, 1983

TABLE 4.1-2

REACTOR COOLANT SYSTEM DESIGN PRESSURE SETTINGS

	Pressure, psig Unit 1	Pressure, psig 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	≥ 1865	≥ 1950
Low Pressure Alarm	2135	2135
Pressurizer Pressure Low - Safety Injection	≥ 1815	≥ 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 Reactor Coolant System pressure drops below 390 psig for Unit 1, 425 psig for Unit 2, a safeguard circuit is manually switched on which allows opening of that Unit's two Power Relief Valves at 400 psig for Unit 1, 435 psig for Unit 2, for low temperature overpressure protection of the Reactor Vessel.

TABLE 4.1-5

STEAM GENERATOR DESIGN DATA*

Number of Steam Generators	4	
Design Pressure, Reactor Coolant/Steam, psig	2485/1085	
Reactor Coolant Hydrostatic Test Pressure (tube side-cold), psig	3107	
Design temperature, Reactor Coolant/Steam, °F	650/600	
Reactor Coolant Flow, lb/hr	33.9×10^6	
Total Heat Transfer Surface Area, ft^2	51,500	
Primary Side:	Unit 1	Unit 2
Heat Transfer Rate (per unit), Btu/hr	2773×10^6	2903×10^6
Coolant Inlet Temperature, °F	599.3	606.35
Coolant Outlet Temperature, °F	536.3	541.27
Flow Rate, (per unit), lb/hr	33.9×10^6	33.7×10^6
Pressure loss, psi	31.4	31.4
Heat Transfer Area, ft^2	51,500	51,500
Secondary Side:		
Steam Temperature at full power, °F	512.1	521.1
Steam Flow, lb/hr	3.53×10^6	3.685×10^6
Steam Pressure at full power, psia	758	820
Maximum moisture carryover, wt %	0.25	0.25
Feedwater Temperature at No. 6 Heater Outlet	436.5	431.3
Fouling Factor, hr-ft^2-°F/Btu	0.0002	0.00005
Overall Height, ft-in.	67-8	
Shell OD, upper/lower, in.	175.75/135	
Number of U-tubes	3388	
U-tube outer Diameter, in.	0.875	
Tube Wall Thickness, (minimum), in.	0.050	
Number of manways/ID, in.	4/16	
Number of handholes/ID, in.	2/6	

*Quantities are for each steam generator

TABLE 4.1-5 (cont'd.)

STEAM GENERATOR DESIGN DATA*

	Rated Load	No Load
Reactor Coolant Water Volume, ft3	1080	1080
Primary Side Fluid Heat Content, Btu	28.7×10^6	27.7×10^6
Secondary Side Water Volume, ft^3	1837	3524
Secondary Side Steam Volume, ft3	4030	2344
Secondary Side Fluid Heat Content, Btu	5.738 x 107	9.628 x 107

*Quantities are for each steam generator

4.2.9 REACTOR COOLANT FLOW MEASUREMENTS.

Elbow taps are used in the Reactor Coolant System as an instrument device that indicates the status of the reactor coolant flow[4]. The basic function of this device is to provide information as to whether or not a reduction in flow rate has occurred. The correlation between flow reduction and elbow tap read out has been well established by the following equation: $\frac{\Delta P}{\Delta P_o} = \left(\frac{\omega}{\omega_o}\right)^2$, where ΔP_o is the referenced pressure differential with the corresponding referenced flow rate ω_o and ΔP is the pressure differential with the corresponding flow rate ω. The full flow reference point is established during initial plant startup. The low flow trip point is then established by extrapolating along the correlation curve. The technique has been well established in providing core protection against low coolant flow in Westinghouse PWR plants. The expected absolute accuracy of the channel is within ± 10% and field results have shown the repeatability of the trip point to be within ± 1%. The analysis of the loss of flow transient presented in Sub-Chapter 14.1 assumes instrumentation error of ± 3%.

4.2.9.1 Reactor Coolant Margin To Saturation

A digital subcooling monitor is provided to display in the control room either the temperature or pressure margin available for the sub-cooled operating condition below the corresponding saturation pressure or saturation temperature. The device selects the highest temperature reading from 8 core exit thermocouples and 8 hot and cold leg RTD's, and the lowest pressure reading from two wide range pressure sensors, and then calculates the corresponding saturation conditions, and displays the available margin of subcooling below saturation, in either temperature (°F) or pressure (psi).

The plant computer is also used to display the margin of subcooling temperature (°F) on an analog trending device in the control room. The computer uses the lowest pressure reading from two wide range pressure sensors and any of four different temperatures, selected by the operator and derived as follows: 1. Hottest incore thermocouple, 2. Average of the incore thermocouples (excluding hottest and coldest), 3. Hottest hot or cold leg RTD, 4. Average of the RTD's.

REFERENCE, SECTION 4.2

1. "Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems WCAP 7735 (Westinghouse Class 3), July 1971."

2. Ernest L. Robinson, "Bursting Tests of Steam-Turbine Disk Wheels", Transactions of the A.S.M.E., July, 1944.

3. "Application of the Griffith-Irwin Theory of Crack Propagation to the Bursting Behavior of Disks, Including Analytical and Experimental Studies", by D. H. Winne and B. M. Wundt, ASME, December 1, 1957.

4. J. W. Murdock, "Performance Characteristic of Elbow Flowmeters", Transactions of the ASME, September, 1964.

5.1 GENERAL DESIGN CRITERIA

Below are listed those general criteria applicable to containment that governs the design of the plant, followed by those criteria that relate specifically to the containment system.

5.1.1 GENERAL CRITERIA

Quality Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or to the mitigation of the consequences, of nuclear accidents which could cause undue risk to the health and safety of the public shall be identified and then designed, fabricated, and erected to quality standards that reflect the importance of the safety function to be performed. Where generally recognized codes and standards pertaining to design, materials, fabrication and inspection are used, they shall be identified. Where adherence to such codes or standards does not suffice to assure a quality product in keeping with the safety function, they shall be supplemented or modified as necessary. Quality assurance programs test procedures, and inspection acceptance criteria to be used shall be identified. An indication of the applicability of codes, standards, quality assurance programs, test procedures, and inspection acceptance criteria used is given. Where such items are not covered by applicable codes and standards, a showing of adequacy is required.

The reactor containment system is essential to the protection of the health and safety of the public. Consequently, this containment was designed, fabricated and erected to quality standards that reflect its importance.

Quality standards governing the design, selection of materials, fabrication and inspection of the containment system conform to the applicable provisions of recognized codes and good nuclear practice. The reinforced concrete structure was designed in accordance with the applicable portions of codes ACI-318-63 and ACI-301-66. Quality assurance programs, comprising test procedures and acceptance standards used, are identified in Section 5.2.2. The applicability of codes, tests standards and other quality assurance programs, including acceptance criteria, are also discussed in this section.

Performance Standards

Criterion: Those systems and components of reactor facilities which are essential to the prevention, or to the mitigation of the consequences of nuclear accidents which could cause undue risk to the health and safety of the public shall be designed, fabricated and erected to performance standards that will enable such systems and components to withstand, without undue risk to the health and safety of the public, the forces that might reasonably be imposed by the occurrence of an extraordinary natural phenomenon such as earthquake, tornado, flooding condition, high wind or heavy ice. The design bases so established shall reflect: (a) appropriate consideration of the most severe of these natural phenomena that have been officially recorded for the site and the surrounding area and (b) an appropriate margin for withstanding forces greater than those recorded to reflect uncertainties about the historical data and their suitability as a basis for design.

All components and supporting structures of the reactor containment were designed so that they would sustain no loss of function in the event of maximum conceivable ground acceleration acting in the horizontal and vertical directions simultaneously. The dynamic response of the structure is based on appropriate spectral characteristics of the site foundation. Damping of the foundation and structure was included in the design analysis. Other applicable natural phenomena which were considered in the design were flooding conditions, seiches and tornados.

Fire Protection

Criterion: A reactor facility shall be designed to ensure that the probability of events such as fires and explosions, and the potential consequences of such events, will not result in undue risk to the health and safety of the public. Noncombustible and fire resistant materials shall be used throughout the facility, wherever necessary, to preclude such

5.2 CONTAINMENT STRUCTURE

5.2.1 DESIGN CRITERIA

Reactor Containment

Criterion: The containment structure shall be designed (a) to sustain without undue risk to the health and safety of the public the initial effects of gross equipment features, such as a large reactor coolant pipe break, without loss of required integrity, and (b) together with other engineered safety features as may be necessary, to retain for as long as the situation requires the functional capability of the containment to the extent necessary to avoid undue risk to the health and safety of the public.

The reactor containment is a reinforced concrete structure consisting of a vertical cylinder, hemispherical dome and flat base. The interior is divided into three volumes, a lower volume which houses the reactor and Reactor Coolant System, an intermediate volume housing the energy absorbing ice bed in which steam is condensed and an upper volume which accommodates the air displaced from the other two volumes during a loss-of-coolant accident. The condensation of steam in the ice bed limits the containment pressure to values substantially below those for a comparable dry-type containment under the same conditions.

The ice condenser containment, together with the containment spray system, provides the functional capability of containment for as long as necessary following an accident. The design pressure of the containment exceeds the peak pressure occurring as the result of the complete blowdown of the reactor coolant through any rupture of the Reactor Coolant System up to and including the hypothetical double-ended severance of a reactor coolant pipe. The design pressure is not exceeded during subsequent long-term pressure transients resulting from the combined effects of heat sources such as residual heat and metal-water reaction with operation of one train of the emergency core cooling and containment spray systems.

All piping systems which penetrate the containment are anchored at the containment wall. The penetrations for the main steam, feedwater, blowdown and samples lines are designed so that the containment is not breeched due to a hypothesized pipe rupture. The core pipe capability in bending is assumed to be limited to its plastic moment capability based upon the yield strength of the pipe material multiplied by a suitable factor. The factors used were as follows:

2.5 For Stainless Steel Core pipes.

1.65 For Carbon Steel core pipes.

Anchors are designed to withstand the thrust, moment, and torque resulting from a hypothesized rupture of the attached pipe.

Isolation valves are supported to withstand, without impairment of valve operability, loadings including those from maximum potential seismic conditions.

Reactor Containment Design Basis

Criterion: The reactor containment structure, including openings and penetrations, and any necessary containment heat removal systems, shall be designed so that the leakage of radioactive materials from the containment structure under conditions of pressure and temperature resulting from the largest credible energy release following a loss-of-coolant accident, including the calculated energy from metal-water or other chemical reactions that could occur as a consequence of failure of any single active component in the emergency core cooling system will not result in undue risk to the health and safety of the public.

The reactor containment structure and penetrations, with the aid of containment heat removal systems including the ice bed, are designed to limit below 10 CFR 100 values the leakage of radioactive fission products from the containment under those conditions that would result from the largest credible energy release following a loss-of-coolant accident, including a margin to cover the effects of metal-water reaction or other undefined energy sources.

P_1 = 16 psi (in fan-accumulator room due to main steam break)

T = Temperature gradient through the concrete and liner under operating conditions

T' = Temperature gradient through the concrete wall associated with 1.5 times design pressure (18 psi)

T" = Temperature gradient through the concrete wall associated with 1.25 times design pressure (15 psi)

T''' = Temperature gradient through the concrete wall associated with 1.0 times design pressure (12 psi)

TL' = Temperature in the liner associated with an accident pressure of 1.5 times design pressure (18 psi)

TL" = Temperature in the liner associated with an accident pressure of 1.25 times design pressure (15 psi)

TL'''= Temperature in the liner associated with a pressure of 1.0 times design pressure (12 psi)

TL = Temperature in the liner (320°F) associated with 1.5 times main steam break design pressure (1.5 x 16 psi) due to fan-accumulator room main steam line break.

TT = Temperature gradient through the concrete and liner under test conditions

E = Operating basis earthquake

E' = Design basis earthquake

W = Wind load

W' = Tornado

(p) = 3 psi differential due to ambient pressure drop due to tornado

U.P. = Unsymmetrical pressure of 8 psi

Load condition (a) indicates that the containment has the capacity to remain elastic and withstand loads at least 50 percent greater than those calculated for the postulated loss-of-coolant accident alone.

Results of the analysis using load conditions (b) and (c) indicate that the containment has the capacity to remain elastic and withstand loadings at least 25 percent greater than those calculated for the postulated loss-of-coolant accident with a coincident operating basis earthquake or wind loading.

Load condition (d) indicates that the containment has the capacity to remain elastic and withstand loadings at least as great as those calculated for the postulated loss-of-coolant accident with a coincident design basis earthquake, as defined in Chapter 2.

Load condition (e) indicates that the containment has the capacity to remain elastic and withstand loadings at least as great as those calculated for operating temperature with a coincident design tornado.

Load condition (f) indicates that the containment has the capacity to remain elastic and withstand loadings at least as great as those calculated for operating temperature.

Load condition (g) indicates that the containment has the capacity to remain elastic and withstand loadings at least as great as that of an unsymmetical pressure of 8 psi in the ice condenser area coincident with a design basis earthquake.

Load condition (h) is for proof testing.

Load condition (i) indicates that the containment has the capacity to remain elastic and withstand local loadings at least 50 percent greater than that due to a steam line break in the fan-accumulator room.

Scaled load plots for moments, shears, deflection, longitudinal forces, and hoop tension, are shown in Figures 5.2.2-14 to 5.2.2-50. The legend for these plots is shown in Figure 5.2.2-13.

Shrinkage induces cracking in the concrete and an initial compressive stress in the rebar and liner.

Design criteria requires that the concrete carry none of the tensile stresses. Therefore, in the analysis, the concrete that is in tension was considered as being cracked. Where, by analysis, the concrete was shown to have compressive stresses, the effects of the initial shrinkage induced tension would be to reduce the compressive stresses. This effect was not taken into account in the calculations.

The initial compression in the rebars, due to the concrete shrinkage, has not been considered as reducing the rebar tensile stress.

Concrete shrinkage does introduce initial compressive stress in the liner, and this initial stress has been considered to be additive to the liner compression stress due to operating and accident conditions.

The auxiliary building concrete was analyzed by conventional structure analysis techniques (i.e., by structural computer programs or hand computations). If the sections assumed in the analysis were satisfactory, reinforcing was determined in accordance with design method in ACI-318-63. If the section assumed was not correct, the above procedure was repeated until the analysis and design agreed. The effects of temperature stresses were added directly when determining the section capacities.

Equilibrium checks of internal stresses and external loads were made. The computer program used for the analysis and design of the containment structure shell was "The GENSH 5 Multi-Layer Static Shell Program: of the Franklin Institute Research Laboratory in Philadelphia, Pa. The output of this program lists both the external loads at the section desired, and the internal stresses at both surfaces of each layer of the section being analyzed. The equilibrium of the external loads and the internal stresses was also checked at various points by manual computations in order to spot-check the computer output.

All structural components are designed to have a capacity required by the most severe loading combination. The loads resulting from the use of these equations are hereafter termed "factored loads".

The design includes the consideration of both the primary and secondary stresses. The design limit for tension members (that is, the capacity required for the design load) is based upon the yield stress of the reinforcing steel.

No steel reinforcement experiences average strains beyond the yield point at the factored load. The load capacity of the structure, so determined, is reduced by a capacity reduction factor "Ø" which is provided for the possibility that small adverse variations in material strengths, workmanship, dimensions and control, while individually within required tolerances and the limit of good practice, occasionally may combine to result in actual capacity lower than the determined value. For tension members the factor "Ø" is 0.95. The factor "Ø" is 0.90 for flexure and 0.85 for bond and anchorage.

A "Ø" value of 0.75 was used for all Class I structural members carrying loads in shear which were produced by earthquake only. For combinations of earthquake loads with LOCA loads a "Ø" value of 0.85 was used for structural members carrying loads in shear.

The capacity reduction factor of 0.75 for shear, which is more conservative than that required by the ACI code, was used for earthquake load alone in recognition of the fact that the potentially relatively large component of shear load associated with earthquake can be considered dynamically applied thereby justifying some additional conservatism.

The load factors used in the equations of Section 5.2.2.3 make provisions for integrity of the containment structure, by the same philosophy used in the ultimate strength procedure in ACI 318-63.

Because of the refinement of analysis and the restrictions on construction procedures, the load factors in the design primarily provide for a safety margin on the load assumptions. The load factors utilized in this criterion are based upon the load factor concept employed in Part IV-B of "Structural Analysis and Proportioning of Members-Ultimate Strength Design" of ACI 318-63. The load factor applied to earthquake or wind load is consistent with that utilized in ACI 318-63. The reduction in the load factor applied to the pressure and thermal loads, when the design earthquake or maximum wind velocity is experienced is also consistent with ACI 318-63. Therefore applicable provisions of "Building Code Requirements for Reinforced Concrete" ACI 318-63 are utilized.

5.2.2.4 Divider Barrier

It is an essential requirement of the ice condenser containment that the steam and air flowing from the lower containment compartment in the event of a failure of a pipe in the Reactor Coolant System, be routed to the upper compartment via the ice bed. To accomplish this, a structural barrier within the containment vessel separates the lower and upper containment compartments. This divider barrier includes the walls of the ice compartment, the upper deck, the compartments enclosing the upper portion of the steam generators and pressurizer, the gate separating the reactor cavity from the refueling canal, and portions of the walls of the refueling canal. The interior wall of the ice compartment also serves as the crane support wall.

It is not necessary to apply a vapor barrier to the exterior surface of the containment wall for the height of the ice condenser compartment. The exterior wall of the ice condenser is separated from the structural concrete and is composed of insulated wall panels which form a complete sheet metal vapor barrier for the refrigerated ice condenser compartment. This vapor barrier is the exterior surface of the insulation and is the warm side.

The operating deck portion of this barrier is supported at its outer radius by short reinforced concrete columns extending above the lower crane wall. The deck is supported at its inner radius by the reinforced concrete primary shielding wall around the reactor. The removable central portion of this deck spans the reactor cavity above the reactor vessel. The operating deck includes hatches above the reactor coolant pumps.

Other portions of the divider barrier are penetrated by hatches for general access and materials handling. The hatch covers and the bulkhead walls between the reactor cavity and the refueling canal are designed to limit post-accident leakage between the lower and upper containment volumes.

The divider barrier between the upper and lower containment compartment is designed to carry the differential pressure between the lower and upper compartments during the postulated loss-of-coolant accident under factored load conditions (a), (b) and (d). The portions of the divider barrier which enclose confined spaces in which a pipe rupture could occur; such as, the steam generator compartments and the slab above the reactor vessel are designed with consideration for differential pressures as a function of available relief area. In addition, the barrier is designed to withstand impact from credible missiles and the effects of fluid jets and pipe whip (where they could occur) without loss of function. For these conditions localized plastic action is accepted and structural ductility is considered in determining equivalent static loads.

Figures 5.2.2-51 to 5.2.2-55A indicate the elements of the divider barrier, the reinforcing used in the barrier elements and the pressure loading applied to these elements.

Analysis of the crane wall, operating deck, steam generator and pressurizer enclosures segments of the barrier structure was by computer. A check was made by manual calculations.

The manual analysis check for the enclosures and the crane wall is based on "Theory of Plates and Shells"* and "Beams on Elastic Foundations" **

The manual analysis check for the operating deck assumes a representative restrained one way slab strip.

The Reactor Access Opening Cover and the Bulkhead are designed manually considering them to be simply supported one-way slabs.

Reinforcing and concrete sections were designed using "Ultimate Strength Design" criteria for the accident conditions and "Working Stress Design" criteria for the operating conditions.

Loading combinations were those listed in section 5.2.2.3.

Additional loads are as follows:

1) An internal design pressure of 20 psi in the steam generator enclosure factored in accordance with the equations of sect. 5.2.2.3.

2) An internal design pressure of 15 psi in the pressurizer enclosure factored in accordance with the equations of sect. 5.2.2.3.

* "Theory of Plates and Shells" by Timoshenko, published by McGraw-Hill Book Co. 1940.

** "Beams on Elastic Foundations" by Hetenyi, published by The University of Michigan Press, Ann Arbor, Michigan 1955.

3) An 8 psi external design pressure on the upper compartment crane wall in the ice condenser area.

4) Thermal load.

5) Jet impingement force.

6) Missile impact force.

7) Pipe reactions and thrust force on the main steam line support anchors.

8) Steam generator lateral support loads due to earthquake and loss of coolant accident or main steam line break.

9) A pressure differential across the operating deck of 12 psi.

SHRINKAGE

The effects of shrinkage are to impose tensile stresses in the concrete and compressive stresses in the reinforcing steel. For a volume/surface area ratio of 12, a conservative value of shrinkage strain equal to 200×10^{-6} (in/in), based on the Matlock and Hansen Graph (see reference page 5.2-101) was used. The computed tensile stresses in the concrete are 56 psi in the crane wall and 34 psi in the steam generator and pressurizer enclosures. These values were added to the stress values determined in the loading combinations.

TEMPERATURE

Two factors are considered in the calculation of thermal stresses.

(1) A maximum thermal gradient of 20°F across the thickness gives a tensile stress of 174 psi in the 3' -0" concrete cross-section and 168 psi in the 2' -0" concrete cross-section of the barrier structure.

(2) A maximum mean thermal rise of 50°F is considered axially in the hoop and meridional directions of the barrier structure.

The results from (1) and (2) are superimposed onto the values determined in the loading combinations.

Accident thermal load increments are of too short a duration to completely penetrate the concrete thickness during blow-down interval. Thermal gradient values used at operating conditions are conservative and do not take into consideration the temperature drop at the skin surfaces of the wall. If this were done the values could be reduced.

JET IMPACT

The unattenuated steam blast from a main steam pipe break inside the steam generator enclosure was considered. The failure of the main steam pipe may occur at any place in the pipe system. The mode of failure is considered as either a longitudinal split or a double-ended rupture. The manual calculation of stresses was based on elastic analysis, assuming interaction of two major types of elements.

a) Vertical strips of annulus sidewall acting as beams supported at the top and bottom.

b) Circular hoops, 1 ft. wide, around the steam generator enclosure wall.

These two elements act together to resist the worst accident combining the jet impact and the instantaneous internal pressure (20 psi). The computed maximum tensile stresses in the steam generator enclosure wall if resisted only by meridional bending elements are 259 psi in the concrete and 2,500 psi in the reinforcing. The combined action of elements in both hoop and meridional direction results in much lower stresses.

MISSILES

The generation of missiles from the reactor control rod drive mechanism was considered in the design of the reactor access opening cover and the primary shield wall. The concrete was analyzed for missile penetration by the modified Petry formula (Ref. The Bureau of Yards and Docks of the U. S. Navy "Designing Bomb-Resistant Structures").

The maximum possible depth of penetration was found to be 0.66 inches in either the 4' -0" thickness of the reactor access opening cover or the primary shield wall. The minimum margin of safety is therefore, F. S. = 48/0.66 = 72 against full penetration.

See FSAR Chapter 14 Safety Analysis, Section 14.3.4, for a discussion of leakage through the barrier. Sensitivity coefficient leakage has been found to be .081 psi in containment pressure increase per ft^2 of deck leakage. An upper bound for maximum size break in event of a DBA would be approximately ten times the design bypass area of 5 sq. ft. This is arrived at by taking the difference between containment design pressure and maximum pressure due to DBA and the above coefficient.

5.2.2.5 Structural Materials

The design of the containment vessel structure is based on specifications giving acceptable limitations of physical and chemical properties for the structural materials used. For certain materials, Indiana & Michigan Electric Company performed physical and/or chemical tests prior to selecting such materials for this project.

The organization, responsibilities, and general provisions for Quality Control are described in Sub-Chapter 1.7. The specific quality control procedures imposed by Specification requirements are outlined herein.

CONCRETE

Structural concrete work has been performed in accordance with "Building Code Requirements for Reinforced Concrete" (ACI 318-63) and "Specifications for Structural Concrete for Buildings" (ACI 301-66).

To supplement the requirements set forth by ACI Standards and Codes, The Bureau of Reclamation Concrete Manual was also used. Compressive strength testing has been performed in accord with ACI 214-65 and ASTM C-39. All concrete used in class I structures has a minimum compressive strength of 3,500 psi at 28 days with or without fly ash.

ACI-301, "Specification for Structural Concrete for Buildings", has been followed in the construction of the Donald C. Cook Nuclear Plant, except where the project specifications have provided detailed instructions.

Portland Cement conforms to Specification for Portland cement, ASTM C-150, Type I.

In addition to the test performed by the cement supplier (those tests specified in the "Specification for Portland Cement" ASTM C150) the following tests were performed by the Sporn Materials Laboratory (now known as the AEP Civil Engineering Laboratory) for I & M Electric Co. to assure that the cement conforms to the ASTM C150 Specification.

a) ASTM C 114 - Standard Methods for Chemical Analysis of Hydraulic Cement.

b) ASTM C 204 - Standard Method of Test for Fineness of Portland Cement by Air Permeability Apparatus.

c) ASTM A 191 - Standard Method of Test for time setting of Hydraulic Cement by the Vicat Needle.

All concrete contains a maximum of 30% (by weight) fly ash which conforms to the "Specification for Fly Ash" ASTM C 618.

Prior to the selection of fly ash a series of tests were conducted by the Sporn Materials Laboratory (now known as the AEP Civil Engineering Laboratory) for determining its chemical properties, thus assuring a high quality concrete. In addition, the Laboratory had performed periodic tests on the fly ash to ensure that its properties are within the limits set forth in ASTM Specification C 618.

Concrete aggregates conform to ASTM Specification C-33-64.

The Course aggregate used in this project was crushed dolomite and it was graded to the following limits:

<u>Sieve Size</u>

Square Openings	Total % Passing by Weight
1"	85 - 95
3/4"	30 - 65
3/8"	0 - 10

Fine aggregate (sand) was obtained locally and has a fineness modulus between 2.7 and 3.0. Fine aggregate was graded within the following limits.

Sieve Size	Total % Passing by Weight
# 4	95 - 100
# 8	85 - 95
# 16	60 - 75
# 30	35 - 60
# 50	10 - 30
# 100	2 - 8

The type and size of aggregate, slump, and additives had been established to ensure high concrete quality with the specified strength and to mimimized shrinkage and creep. Neither calcium or any admixtures containing calcium chloride or other chlorides, sulphides, or nitrides were used.

Mixing water was controlled by periodic testing to ensure that it did not contain more than 1000 ppm of the above chemical constituents.

Purpose of Concrete Admixtures

All structural concrete contains a water reducing admixture and an air entraining admixture meeting ASTM specifications C-494 and C-260-67 respectively. "Placewel R" was selected as the water reducing agent and "Aircon Double Strength" as the air entraining agent, both manufactured by Union Carbide Corporation. Dosage requirements for the basic design mixes was determined in accordance with Manufacturer's recommendations and trial mixes performed by the concrete laboratory. "Placewel R" is used primarily to reduce water requirements in the mixes and thus reduce the cement content without sacrificing workability or strength. "Aircon" was used primarily to increase the durability of the concrete.

Inspection and Surveillance

The following quality control measures outlined below apply to structural concrete.

Pre-Construction Tests

Prior to commencing concrete work for this project, the AEP Sporn Materials Laboratory (now known as the AEP Civil Engineering Laboratory) conducted a series of tests on different trial mixes (using the same materials selected for this project) to determine the mix proportions necessary to produce concrete conforming to the strength requirements specified. The majority of the concrete compression tests for these trial mixes showed a 7-day strength equal or greater than that expected at 28-days.

The methods used for sampling, making, curing and testing concrete specimens were in accordance with the following ASTM Standards.

a. ASTM C-192-66 - "Standard Method of Making and Curing Concrete and flexure Test Specimens in Laboratory."

b. ASTM C-34-64" Standard Method of Test of Compressive Strength of Molded Concrete Cylinders."

c. ASTM C-172-54 "Standard Method of Sampling Fresh Concrete".

d. ASTM D-1-65 " Standard Method of making and curing concrete compression and Flexure Test Specimens in the Field".

Field Materials Testing Laboratory

To monitor Quality Control on construction materials, I&ME Co. established a field testing Laboratory which was under the direct control of the Sporn Materials Laboratory (AEP C. E. Laboratory). The field testing Laboratory was manned by competent personnel experienced in the testing of construction materials.

Some of the tests conducted by the field Laboratory were:

a) Testing of coarse and fine aggregates.
b) Testing of concrete cylinders.
c) Concrete slump.
d) Air entrainment.
e) Reinforcing steel.

During testing operations the testing laboratory assigned an inspector at the batch plant to monitor the mix proportions of each batch of concrete produced by the batch plant. The concrete batch plant utilized for this project conformed in all respects, including provisions for storage and precision of measurements, with the "Standard Specifications for Ready Mixed Concrete" ASTM C 94-68.

The batch plant inspector tested, periodically, the mix ingredients and ensured that a tape record was provided for each batch, documenting the time loaded, actual proportions of the mix, amount of concrete, concrete design strength, destination as to portion of structure, identification of transit mixer, and reading of revolution counters at first addition of water.

Whenever ready-mixed concrete was required, it was mixed and transported in accordance with "Specification for Ready-Mixed Concrete, "ASTM C94-68. The minimum amount of mixing in truck mixers, loaded to maximum capacity, was 70 revolutions of drum or blades after all of the ingredients, including water, were in the mixer. The maximum number of revolutions at mixing speed was 100. Records were maintained as to the time and reading of the revolution counter when concrete was discharged.

Inspectors at the construction site, inspect reinforcing and concrete placement and curing.

For Class I structures (containment vessel, auxiliary building and other structures) test cylinders and concrete compression tests were taken based on the following schedule:

Concrete Poured in cu. yds.	Samples taken
0-100	1 for each 100 cu. yds.
100-1000	1 for each 500 cu. yds.
1000-2000	1 for each 700 cu. yds
2000 and over	1 for each 1000 cu. yds.

A sample consists of 2 cylinders to be tested at each of 3.7 and 28 days.

For every mix design, and prior to the production of structural concrete, five (5) slump tests were made and an average value was established. This value was within the range of 3" to 5". Slump tests were also made at the time concrete test cylinders were cast. They were also made at the batch plant and at least each hour during pouring time.

During mass concrete operations, for obtaining the desired slump, the batch plant operator "holds back" a portion of the theoretical quantity of water, as determined by the approved design mix. As a result the concrete produced is of low slump since a portion of the full amount of water specified in the mix design was "held back".

The slump of the concrete was determined by means of an ammeter attached to the mixer drum motor. If the ammeter reading indicated low slump more water was added to bring the slump up to within the specified range of 3" to 5". The added amount of water was recorded.

The amount of "hold back" water was estimated based on the moisture content of the sand. The Cook Plant practice was that no water was added to the concrete after it left the batch plant.

In addition to the slump control outlined above, at least two manual slumps were taken whenever test cylinders were cast. Furthermore, whenever a large pour was being made (500 cu. yds. or greater) one slump test was taken at the batch plant every hour for the duration of the pour.

Over the course of the project the average compressive strength of the 28 day cylinders either met or exceeded the specified compressive strength of 3500 psi.

All slump tests were conducted in accordance with the "Method of Test for Slump of Portland Cement Concrete" ASTM Specification C 143-58. Batch rejection was based on deviation from specified slump specifications. Pour removal would be based on an engineering analysis of core cylinder tests that would be instituted following the failure of strength cylinder tests to meet 90 percent of the specified average strength.

Concrete samples for the Cook Plant were taken from the transport trucks at the site concrete laboratory which was located adjacent to the mixing plant. This is in conformance with ACI-214, section on "Tests and Specimens Required".

Reinforcing Steel - Material and Specification

Reinforcing steel is deformed new billet steel bars conforming to the requirements of "Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement ASTM Designation A 615-68". This steel has a minimum yield strength of 40,000 psi.

Rebar Inspection and Testing

Certified reports of chemical and physical test performed on the reinforcing steel are submitted to the Engineer by the supplier. These tests conform to the requirements of "Standard Specification for Deformed Billet-Steel Bars for Concrete Reinforcement" ASTM Designation A615-68.

In order to assure that reinforcing steel met appropriate specifications, samples of rebar delivered to the job site were selected and tested to confirm compliance with the specified physical requirements and for certification of mill test reports.

The selection of the specimens was as follows:

Two specimens were taken for each heat of material. No samples selected included the end 12 inches of any bar delivered.

Specimens were tested for ultimate strength, yield strength and elongation by Indiana & Michigan Electric Company, prior to fabrication and/or delivery of the reinforcing to the job site. If any of these specimens failed to meet the requirements of the applicable specification for ultimate strength, yield strength or elongation, the heat of steel was resampled, this time selecting four specimens instead of two as were required originally. If any of these specimens failed to meet the requirements of the applicable specification for ultimate strength, yield strength or elongation, the entire heat was rejected.

All reinforcing was kept separated by size and heat and tagged with the manufacturer's identification number. This identification was maintained at least until the heat of steel met the aforementioned requirements.

To insure that only the specified reinforcing steel was received, the mill test reports for each shipment were checked against the mill test reports sent to the job site with the test specimens.

Only two grades of rebars were used during the entire project, both of which met the requirements as stated. This eliminated the possibility of substitution of an inferior grade of steel during erection.

Generally, grade 40 rebars were used during erection of all structures. If grade 40 was not available, grade 60 rebars (which are superior in strength) were used. However, only a very limited amount of grade 60 rebars were used and the stresses were kept to grade 40 allowables.

Since the low operating temperature of the ice condenser would not have adverse affects on the reinforcing steel with respect to its physical properties, tests for determining the NDTT properties of the material were not required.

<u>Reinforcing Steel Splices - Specifications</u>

The main load carrying reinforcement is spliced by the Cadweld process or lap spliced as noted in Section 5.2.2. These Cadweld splices are designed to develop the average minimum ultimate tensile strength of the ASTM grade of reinforcing bars being spliced, with no splice falling below 125% yield. Lap splicing will be permitted for secondary or flexural load carrying bars up to and including No. 11. Lapped splices where used, have followed provisions of ACI Code 318-63, Section 805.

Cadweld splice staggers in the containment structure have been maintained at 6 ft. minimum between splices in adjacent bars for the foundation mat and 2 ft. minimum between splices in adjacent bars for the containment wall and interior structure. No tack welding was permitted.

Installation Procedure - Cadwelds

In addition to the manufacturer's splicing procedure, the following procedures were observed:

a. Cadweld splice sleeves and powder were stored in such a manner as to avoid wetting or soaking from snow and rain, to prevent rusting of splice sleeves and to prevent wetting of powder prior to field usage. When being used the powder was protected from water and moisture by water tight containers.

b. Rebar ends to be spliced were wire-brushed, by means of a powered wire brush, to remove all loose mill scale, red rust and adhering concrete.

c. Rebar ends which were wet, grease or mud covered were dried with a torch before wire brushing.

d. Rebar ends which were painted had the paint burned off by a torch before wire brushing.

e. A line was marked 12" ± 1/4" from the end of the bar with a paint marker. This line was used as a reference point to insure that the bar ends were centered in the splice sleeve.

f. Clean rebar ends were heated, to assure complete absence of moisture, immediately before the splice sleeve was placed into final splicing position.

g. With all packing materials, equipment and graphite pouring basin in position, the splice sleeve was heated externally, until it was warm to the touch, when the temperature was below 32°F or the humidity was above 65%.

Prior to any production splicing, each operator and foreman or supervisor, was instructed by a representative of the manufacturer. Each operator was required to prepare and have tested 3 splices for each of the positions to be used in production work (horizontal, vertical and diagonal). An operator was considered qualified if all three specimens for each position passed the visual and the tensile test performed by Owner's site personnel, qualified in cadweld splicing. A list of qualified operators and their qualified test results is maintained at job site. A manufacturer's representative of Cadweld was present for at least the first one hundred (100) production splices to verify that proper procedures were being used and quality splices obtained.

Inspection and Testing - Cadwelds

The Cadweld splice acceptance procedures used were those of the manufacturer. All completed splices were visually inspected at both ends of the splice sleeve and at the tap hole in the center of the splice sleeve. Sound, nonporous filler metal had to be visible at both ends of the splice sleeve and at the tap hole in the center of splice sleeve for the splice to be accepted. Filler metal recessed 1/4" from the end of the sleeve, due to the packing material, was not considered to be a poor fill.

Randomly selected splices for each crew and position* were tensile* tested. Selected splices, excluding curved rebars of containment bottom slab and dome with radius less than 57'6", were tensile tested by Applicant's Testing Laboratory, in accordance with the following schedule for each crew, position, bar size and grade of bar.

One (1) production splice out of the first ten splices.

Two (2) production and two (2) sister splices out of the next 100 splices

*Specifications require that no splice in the test series shall have a tensile value below 125% of the specified yield point stress of that grade of reinforcing bar to which it is being applied.

One (1) production and two (2) sister splices out of the next and subsequent 100 splices.

Sister splices, where used, were made with test bars of 3 feet in length, spliced in sequence with production bars.

No reinforcing steel splices were checked by non-destructive inspection methods.

. <u>Liner and Anchors</u>

<u>Materials and Specifications</u>

The steel for the liner and attachments conform, where applicable, to:

a) Specification for Low Carbon - High Manganese Normalized steel with Fine Grain Structure" ASTM A-442-66 Grade 60. The liner plate thickness is 1/4" on the bottom and 3/8" on the shell and dome.

b) "Specification for structural steel" ASTM A36-67 for rolled sections including weld channels and stiffeners.

c) The anchorages for the containment liner consist of structural angles conforming to ASTM A 36 Specification and L-shape Nelson Studs (3/8" dia). These studs conform to the requirements of ASTM A-108-69T "Low Carbon Steel".

<u>Inspection and Tests - Anchors</u>

To confirm the structural integrity of the Nelson stud to plate weldment, at the beginning of each day, each welder attached at least one test stud which was tested by bending the stud approximately 45 degrees toward the face of the plate. Whenever failure occurred in the weld, the welding procedure and/or technique was reviewed and corrected, and two successive studs were successfully welded and tested before further studs required by the design were welded to the liner plate. The test studs were allowed to remain in place, but were not considered as part of the regular stud pattern required by the design. All stud welds were visually inspected. Any stud on which a full 360 degree weld was not obtained was removed and replaced by a new stud.

Inspections and Tests - Liner

ASTM standard test procedures were employed for liner plate to ascertain compliance with ASTM A 442-66 Specification. Certified copies of mill test reports describing the chemical and physical properties of the steel were submitted to I&M Electric Company for approval. Test for qualifying welding procedures and welders were performed by the fabricator and monitored by I & M Electric Co. The liner plate material was tested (one test for each heat of steel) to determine its Nil Ductility Transition Temperature (NDTT). These tests were conducted in accordance with the Naval Research Laboratory's Report NRL 6300 on Drop-Weight Tear Test. The tests were conducted at a maximum temperature of 30°F below the minimum service temperature of 0°F. In addition, the plates were impact tested, by the liner fabricator, in accordance with the applicable sections of Paragraph N330, Section III of the ASME Boiler and Pressure Vessel Code, at the same temperature as the Drop-weight tear test (-30°F).

Quality Control measures for welding and weld testing:

All welding electrodes used for the Donald C. Cook Nuclear Plant were kept in "holding ovens" at a temperature of 150°F. Welding electrodes were issued by the job foreman to welders as required.

However, no welding electrodes were allowed to be used if wet or if they had been removed from the holding ovens for more than four (4) hours.

a. All welders and welding procedures were qualified in strict accordance with the requirements of Part A Section IX of the ASME Code (1968).

b. All welds in the bottom (including the reactor pit and recirculation sump), cylindrical shell and dome liners were tested as follows:

   Complete radiographic testing was done for the first 15 accessible feet of weld made by each welder and position, in accordance with Paragraph UW 51 Section VIII of the ASME Code.

   Spot radiography was done for every 50 feet following that portion of the weld completely tested by radiography, except as noted below.

   Those areas of the liner which were impractical to be radiographed or spot radiographed were tested 100% by the Magnetic Particle Test Method per Section VIII of the ASME Code.

   All liner welds were 100% vacuum box tested.

Upon completion of the non-destructive testing of welds, all welded seams were covered by test channels, which were tested for strength and leakage as follows:

a. The channels were pressurized with air to 50 psig for 15 minutes (strength test).

b. Following the strength test, the channels were pressurized with a 20% by weight of Freon-Air mixture. By means of a halogen leak detector, having a sensitivity of 10^{-7} standard cubic centimeters per second, 100% of the welds were tested for leakage. Furthermore, the weld channel zones (a group of connected channels) were tested at a pressure of 15 psig for two hours with no drop in pressure above acceptable limits taking into account pressure variations due to temperature variation.

The following additional documents were used to supplement the basic document (ASME Section III). Dates of references are the latest edition at the time of order placement.

a. ANSI B16.5 - Steel Pipe Flanges and Flanged Fittings
b. ASME Boiler and Pressure Vessel Code, Sections VIII, IX and II
c. ANSI B16.11 Socket Weld Fittings
d. ASTM Standards
e. American Electric Power Service Corporation Specifications for Nuclear Piping, Piping Materials, Containment Liner
f. Westinghouse Electric Corporation Process Specification 83336KA and Appendices A & B.
g. USAS B31.1 - 1967.

5.2.2.6 Corrosion Protection

The portion of the Containment Building which is below the ground water table (GWT) at approximate elevation 585 has been waterproofed by means of a PVC 40 mil plastic membrane. Realizing the seasonal fluctuations in GWT, the membrane is applied well above the highest known GWT elevation.

In addition, I & M Electric Co. has conducted a series of tests to determine whether or not natural or man-made underground corrosion tendencies were present at the plant site. During these investigations two important factors were considered:

a) The ability of the soil to sustain or accelerate any corrosion cells that might be established.

b) The behavior of any man-made d.c. currents present.

Since it is known that soil electrical resistivity and acidity are good indicators of soil corrosion tendencies, the purpose of the tests was to measure these two parameters. The soil resistance to electrical charges was measured with Vibroground using the four-pin method. This method gives the average soil resistivity from the surface to the pin spacing. Five pin spacings varying from 10 to 50 feet were used at most test locations (for test locations see Figure 5.2-1). The values of 10,000 Ohm-centimeters or less are the values commonly considered conducive to corrosion.

The values measured are listed in Table 5.2-2. Acidity tests were made, where possible, by means of pH paper to determine the chemical aggressiveness of the electrolyte. Very slight acidity was found in the lake water, varying between 6.5 and 6.8. Tests were also made in the immediate plant site area and a pH of 6.5 was noted. These values are close to the neutral pH of 7 which is indicative of a passive environment.

To determine the presence of stray d.c. current, potential drop tests were made at the plant site. These tests indicate no stray currents were present. The area around the plant site was investigated for pipelines under cathodic protection. Two pipelines were found to run roughly parallel to the lakeshore and under cathodic protection with pipe to soil potentials averaging 2.25 volts. No rectifier units were found in a six mile section of these lines and it was concluded that they have no effect in the plant area.

Based on the above investigation it was concluded that the underground environment at the plant site does not promote corrosion. However this does not preclude the possibility that man-made corrosion cells introduced into this environment will not promote corrosion. Realizing this fact, considerable care has been exercised to eliminate from the design all electrically connected dissimilar metals, foreign electrolytes in the vicinity of metals, stray d.c. currents and other corrosion promoting devices.

The exposed surface of the containment liner (vertical cylindrical shell and dome) was coated with Carbozine No. 11 as primer and Phenoline white No. 305 as finish coat. The total thickness is approximately seven (7) mils. The outer surface of the steel is directly in contact with the concrete which provides adequate corrosion protection due to the alkaline properties of the concrete.

For the containment reinforcing a 3 inch cover of concrete was provided. This is approximately 50% greater than that specified by ACI-318 code.

5.2.2.7 Structural Design for Jet Loads

An analysis has been made to summarize the capability of the containment divider barrier and compartments, to withstand the jet force effects of a reactor coolant loop (DBA) or steam line break inside the containment building.

The reactor coolant system is provided with pipe whip restraints. Both circumferential and longitudinal ruptures were considered in the design of this restraint system. Circumferential ruptures were considered at all changes in direction and nozzle junctions in the RCS and connecting systems. Longitudinal ruptures were postulated to occur at selected locations within the reactor coolant pressure boundary.

Four restraints have been provided on each of the four main steam risers as well as two restraints on each steam line immediately before exiting the containment. Restraint cross sections are shown in Fig. 5.2.2-56.

The reactor coolant, main-steam and feedwater lines have been restrained outside and within the steam generator and pressurizer enclosures such that damage to the containment, safeguard systems and an increased severity of a LOCA would not occur from pipe whip or blowdown jet forces.

The jet effects assessed are the result of conditions arising from the following postulated breaks:

Reactor Coolant Piping System

1. Reactor Vessel Outlet Nozzle - Partial Guillotine

2. Reactor Vessel Inlet Nozzle - Partial Guillotine

3. Steam Generator Inlet Nozzle - Guillotine

4. Steam Generator Outlet Nozzle - Guillotine

5. Reactor Coolant Pump Inlet Nozzle - Guillotine

6. Reactor Coolant Pump Outlet Nozzle - Guillotine

7. 50° Elbow on the Intrados - Split

8. Flow Entrance to the 90° Elbow - Guillotine

9. RHR Primary Loop Connection - Guillotine

10. Safety Injection/Primary Coolant Loop Connection - Guillotine

11. Pressurizer Surge/Primary Coolant Loop Connection - Guillotine

12. Loop Closure Weld in Crossover Leg - Guillotine

13. Surge Line Inlet to Pressurizer - Guillotine
(Pressurizer Compartment)

Main Steam Pipe System

14. Main Steam Line Nozzle of Steam Generator - Guillotine
(Steam Generator Compartment)

15. Main Steam Line - Guillotine
(Fan-Accumulator Compartment)

Figs. 5.2.2-56 and 5.2.2-56A illustrate the break locations representing the Westinghouse criteria for break locations in the RCS and the restraints and physical geometry of the structures subject to jets, including the steam generator and pressurizer enclosures, the fan-accumulator rooms, and the operating deck. The crane wall is subjected to reactions from the steam generator snubbers acting as rigid supports during an earthquake and/or DBA. The combined DBA and DBE load in the steam generator enclosure is 1600 KIP's at each reaction point and was factored into the design of the crane wall.

In addition to loading conditions (a) through (i), of Section 5.2.2.3, the containment barrier and associated enclosures were analyzed for jet effects using the following loading conditions:

$$(1.0 \pm .05)\ DL + 1.0F_I + 1.0T' \quad (1)$$

$$(1.0 \pm .05)\ DL + 1.25P + 1.0\ F_S + 1.0T' \quad (2)$$

Where:

F_I = equivalent static jet load effects at the initiation of the break.

F_S = equivalent static jet load effects during the saturated pressure phase, and all other terms as defined in the FSAR.

These equivalent static jet load effects were determined from the time history forcing function at the point of postulated break, and are considered to act on the affected structure, with the following assumptions:

1. Peak response of structure due to initial and saturated jet impingements on divider barrier.

   a. Assuming a ductility factor equal to 3 in regions where moment governs design.

   b. Assuming a ductility factor equal to 1.3 in regions where shear or diagonal tension govern design.

2. Peak response of structure due to initial and saturated jet impingement on internal structure other than the divider barrier.

   a. Assuming a ductility factor equal to 10.0 in region where moment governs design.

   b. Assuming a ductility factor equal to 3.0 in regions where shear or diagonal tension govern design.

In those cases where a calculated time history forcing function defined at the point of a break is not available, such forcing functions shall be conservatively defined as a rectangular pulse with zero rise time and a duration at least ten (10) times the fundamental period of the affected structure. The magnitudes of these forcing functions are:

$F_s = 1.2\ p_i\ A$ and

$F_s = 1.2\ p_s\ A$ where:

p_i = system normal operating pressure of the initiation of the break

p_s = saturation pressure evaluated from the piping pressure response after the postulated break.

A = cross sectional area of the pipe.

Based upon air analysis of piping pressure transients, F_s is taken as 2/3 F_i. For assumed slot failures, the break opening is taken as a length equal to twice the diameter of the pipe and having an area equal to the cross sectional area of the pipe. The jet is assumed to diverge with a solid angle equal to 10° on each side.

For purposes of calculating jet impingement loads, one pipe diameter displacement (i.e. outside edge to outside edge) was assumed unless it was physically impossible to get the displacement, as in the sleeve through the reactor cavity shield. This is a conservative assumption for guillotine breaks of the reactor coolant system and the steam line, since hinges cannot form and the pipes will move laterally away from each other only slightly.

Stress Criteria

The allowable shears for the jet loads were determined by the following formulas:

(a) Peripheral shear (governing in line of punching shear)
Art. 1707, AC1 318-63

$$v_u = \frac{\nu\mu}{b_o d} \qquad \text{(egn. 17-7 of AC1 318-63)}$$

$$v_u = 4\phi \sqrt{f'c}$$

where b_o = periphery of critical section

$$= 2\pi \left(R + \frac{d}{2}\right)$$

where R = radius of jet cone, with other symbols as defined on p. 318-68 of AC1 318-63.

(b) Radial shear - Art. 1701, AC1 318-63

$$v_c \leq 3.5\phi \ f'c \sqrt{\left(1.0 - .002 \frac{N}{Ag}\right)}$$

where N = axial tension

and

$$v_c \leq \phi \left(1.9 \sqrt{f'c} + 2500 \frac{pwVd}{M'}\right) \qquad \text{(egn 17-2 of AC1 318-63)}$$

$\leq 3.5\phi \sqrt{f'c}$

where $M' = M + N \left(\frac{4t-d}{8} \right)$ (egn 17-3 of Acl 318-63)

For shears exceeding the values listed above, web reinforcing was added.

Results of the Analysis

The analysis for the jet loads involves a consideration of the "source" and the "target". Each postulated pipe break was considered as a source and the barrier or compartment internal structures were considered as targets. Each target was analyzed for the effects from each source.

Basically, all targets are protected from jet forces in at least one of the following manners:

a. The source and the target are physically separated or the break orientation is such that the structure is not a target.

b. The energy level of the source is insignificant relative to the target.

c. There is interference between the source and target such that (a) and (d) apply.

d. The target is capable of resisting resulting jet impingement forces.

The targets considered included the operating deck, steam generator and pressurizer enclosures, crane wall, fan-accumulator rooms, missile shield, reactor cavity primary shield, containment wall, and fill slab. Note, that in all cases, the containment wall integrity was not affected by jet impingement. The results of the analysis are presented in Table 5.2-5. Only the primary target for each break is presented, the secondary targets being subjected to much lower forces.

This analysis was conducted in a conservative manner since (1) the break locations considered are more severe than Westinghouse position papers and App. B2 of ANS-20 indicate, (2) no energy dissipation due to distance or turbulent discharge was considered, and (3) the pipe was assumed to displace one pipe diameter where, in actuality, the pipe would not hinge.

5.2.3 VESSEL STRUCTURAL ANALYSIS (STATIC)

Basically three separate containment vessel structural components are analyzed, each in equilibrium with loads acting on it and with constraints occurring at the juncture with other structures. The three structural components are:

a) The hemispherical dome
b) The right cylinder
c) The base mat

Since the thickness of the dome and cylinder are small in comparison with the radius of curvature (cylinder 3.5/57.5 = 1/16.4; dome 2.5/57.5 = 1/23.0), the dome and cylinder were treated as thin-walled shell structures.

All tensile stresses were assumed in the design to be carried by the reinforcing steel. No credit was taken in the design of the shell, for the liner capability to carry tensile, compressive or shear stresses.

Discontinuity stresses occur at changes in section or direction of the containment shell.

The juncture of the cylinder to the dome is a point of discontinuity since the dome and cylinder have different radial stiffnesses under load.

The juncture of the cylinder to the base slab is a point of discontinuity. In the analysis, the cylinder base slab juncture was considered to be a point of infinite rigidity and the cylinder at this point does not expand or rotate under the internal pressure and temperature load conditions.

The containment vessel structure was analyzed in the following manner:

1) The forces due to pressure wind (or tornado), dead load and thermal considerations were determined by thin shell theory following procedures indicated in "Thin Shell Concrete Structures" by D. Billington and "Stresses in Shells" fourth printing by Wilhelm Flugge.

2) The dome and cylinder were initially treated as independent structures and the primary systems solved. The edges of the structures were considered free to displace (translate and rotate). This solution results in membrane stresses.

3) The magnitudes of the edge displacements were determined.

4) The amount of translational and rotational displacement due to unit edge loads at the boundaries were determined.

5) At the joint between dome and cylinder, compatibility was achieved by computing the magnitude of the edge effects

required to eliminate the differential of displacement between the boundary of the dome and that of the cylinder.

6) The results of Step "5" are meridional and hoop stresses which were superimposed on the meridional and hoop membrane stresses resulting from the solution of the primary systems of Step "2" and meridional bending moments.

7) A similar procedure to that of Step "5" was followed for the lower edge of the cylinder, where the cylinder joins the base slab, to achieve compatibility of displacement between this boundary and the base slab.

For additional conservatism in the determination of meridional moments at the points of discontinuity, the concrete was considered to be fully cracked vertically. Poisson's Ratio was not considered and Young's Modulus was taken as the value specified in ACI 318-63, Section 1102; no variation of this value was considered.

The equivalent internal pressure load imposed on the containment shell due to thermal loads was determined considering the fact that commercially available plate could vary by +7 percent or -3 percent from its nominal thickness and that the actual yield point may exceed the minimum yield value by 30 percent.

The equivalent pressure load on the concrete shell, as determined from the liner thermal load, was based on plate being +7 percent greater than nominal thickness and plate stress 30 percent greater than minimum yield.

Unsymmetrical pressure and thermal loadings exist because of various relatively confined areas in the lower compartment and because the ice condenser does not cover the full 360 deg. of the containment structure. The effects of this asymmetry were evaluated.

Analysis of the containment structure for seismic loading was by beam flexure theory. See Appendix F of the Original FSAR. For the seismic analysis a range of shell rigidities was considered to allow for various depths of crack in concrete.

The containment structure was designed by Ultimate Strength methods conforming to the behavior criteria of ACI Code 318-63, Part IV-B - "Structural Analysis and Proportioning of Members - Ultimate Strength Design."

Stress and strain limits conform to ACI-318-63. Capacity reduction factors are as indicated in Section 5.2.2.

Principal reinforcing used in the containment structure has a minimum yield strength of 40,000 psi and a minimum ultimate strength of 70,000 psi. Concrete has a minimum 28-day compressive strength of 3,500 psi. The concrete is considered not to carry any tensile forces.

The radial shear carrying capability of the concrete at each section was evaluated according to the procedure of ACI-318-63, Part IV-B, Section 1701. Where shear reinforcing was required it was considered that all the shear at the section is carried by the shear reinforcing.

Where radial diagonal bars were required, they were not lap spliced with the main vertical or inclined tangential wall bars, but were either bent back and forth between the opposite faces of the wall to form a continuous stirrup or the bars were hooked about the main reinforcing to achieve positive anchorage.

Supplementary reinforcing which was added to accommodate local conditions such as discontinuity stresses was carried a sufficient distance beyond the region where it is required and anchorage was achieved by means of end plates cadwelded to the reinforcing bars.

Dome

The analysis of the hemispherical dome was performed by the superposition of stresses resulting from gravity, accident pressure and thermal loads. In addition, earthquake or wind loading creates both direct and shear stresses in the dome.

Cylinder

Tha analysis of the cylinder was by the superposition of stresses resulting from gravity, pressure and thermal loads, over-turning due to earthquake or wind and shears due to earthquake or wind.

The concrete was reinforced circumferentially using steel hoops and vertically by vertical reinforcing. Tangential shear reinforcing as required to resist shear due to earthquake or wind was placed at 45° to the vertical each side of the vertical.

Although the cylinder wall was considered fixed at the juncture with the base slab, for determination of discontinuity stresses in the thin shell analysis, the effects of base slab edge rotation on the cylinder wall due to elastic subgrade were determined, as were the effects of base slab edge deformation due to accident internal pressure loading. Modification of the cylinder base discontinuity stresses was then made as required.

There is no soil backfill along the lower part of the cylinder wall, therefore, the wall has no elastic restraint due to soil backfill.

The effects of penetrations through the cylinder wall were considered. Penetrations 9 inches or less in diameter do not significantly perturb the reinforcing pattern in the containment wall, therefore no special reinforcing considerations were made at these areas.

For penetrations between 9 inches and approximately 4 feet - 6 inches in diameter, the reinforcing was terminated at the opening. The reinforcing so terminated was anchored by means of end plates cadwelded to the reinforcing bars at the periphery of the penetration, to achieve positive anchorage. Supplemental reinforcing was added in the direction of the main reinforcing, and diagonally, to replace the reinforcing terminated. The area of supplemental reinforcing added is twice that of the area of reinforcing terminated and was placed adjacent to the penetration. The additional reinforcing was extended a sufficient length beyond the area which was considered as significantly affected by stress concentration due to the penetration, so that the additional reinforcing develops its full ultimate strength at ultimate bond stress. In no case is the length of these additional reinforcing bars less than 20 feet. Consideration was also given to hooking the ends of this additional reinforcing to provide positive anchorage, where termination is in a tensile zone.

Openings in the concrete shell greater than approximately 4 feet - 6 inches in diameter are:

a) The equipment hatch
b) The personnel access hatch

Reinforcing of these large openings is by means of a thickened concrete ring beam around each opening.

The external loads applied at the openings are dead load, pressure due to incident conditions, temperature associated with the incident condition and earthquake load. Design combinations considered are essentially the same as for the rest of the cylindrical shell and are considered according to the factored load equations in Section 5.2.2.3. Secondary stresses in the concrete ring beam result from the peripheral forces of the penetration itself; due to the internal pressure of the accident condition, earthquake or tornado. Additionally, secondary stress is induced by the curvature of the ring to match the cylinder.

Analysis of the ring beam and the adjacent area was made by a finite element program by the Franklin Institute Research Laboratories.

Base Mat

The containment Building base mat was analyzed by three independent methods. The STRUDL and GENSHL 5 computer programs were used and manual calculations were made as a check on the computer programs.

The various loads considered were: Dead Load, Soil Reactions, Thermal Loads, Wind Loads, Tornado Loads and Earthquake Load.

Three soil reaction distributions were considered for the static load condition. It was considered most probable that the soil reaction is fairly uniform as indicated in Case I in Figure 5.2.2-57. However, the soil reaction may vary linearly to a condition of maximum bearing pressure under the pit indicated in Case II of Figure 5.2.2-57 as a uniform pressure under the slab and a greater uniform pressure under the pit or it may vary linearly to a condition of maximum bearing pressure at the edges of the slab as indicated in case III of Figure 5.2.2-57.

Lateral soil pressures on the walls of the reactor cavity and the refueling canal were considered in the analysis.

Seismic and wind or tornado conditions cause over-turning moments. The soil reactions for these conditions was directly super imposed onto the static cases as indicated in Case IA, IIA, and IIIA of Figure 5.2.2-57A.

The soil reactions are considered as member loads in the STRUDL computer program. Whereas the "GENSHL 5" program has provision for an elastic foundation material.

The dynamic model of the containment includes a rocking spring below the base slab, and a lateral spring at the base mat elevation. The stiffnesses of these springs, determined from the dynamic soil modulus and the base mat geometry, accounts for the soil below and around the base mat.

The maximum soil pressure component due to the earthquake loads are 2.5 Ksf for the "OBE" and 4.0 Ksf for the "DBE".

The maximum soil pressures for both uniform and non-uniform soil pressure distributions including the DBE pressures are shown in Figures 5.2.2-58 and 5.2.2-58A. Based on the ultimate bearing capacity of the underlying clay stratum of 36 (kips/ft^2), the maximum soil pressure under combined seismic and other appropriate loads of 14.8 ksf provides a factor of safety of 2.4. This factor of safety is conservative since it is based on (unconfined) compression tests of the clay, and since the influence of the sand layer overlying the clay in distributing the load is neglected.

The stresses in the base slab resulting from the internal pressure due to the accident condition were treated separately. These stresses were then added to the stresses previously determined. The edge deformations of the base slab for the accident condition were determined and the modification of the cylinder base discontinuity stresses were made, as stated under "Cylinder Analysis".

The base slab was analyzed for the effects of a temperature gradient of 110°F on the inside surface of the structural concrete adjacent to this stub liner and a 45°F temperature on the outside of the concrete against the soil.

Loading was applied to the base slab in accordance with the factored load equations in Section 5.2.2.3.

Each loading condition for the entire containment structure was calculated separately and the method of superposition was used to obtain the resultant foundation loading and base moments and shears.

The loadings considered for the reactor cavity are:

1. Dead load (concrete)
2. 10 psi external pressure
3. 30 psi internal pressure
4. 65 psi internal pressure
5. Dead load (reactor)
6. Operating thermal load
7. Steam Generator #1 lateral load due to accident - radial
8. Steam Generator #1 lateral load due to accident - tangential
9. Steam Generator #2 lateral load due to accident - radial
10. Steam Generator #2 lateral load due to accident - tangential
11. Reactor lateral load due to Loss of Coolant Accident
12. Seismic lateral load due to Operating Basis Earthquake
13. Seismic lateral load due to Design Basis Earthquake

A) 1 + 4 + 5 + 6

B) 1 + 3 + 5 + 6

C) 1 + (1.2) 2 + 13 + 6 + [(7+8) or (9+10) or 11]

D) 1 + (1.5) 2 + 12 + 6 + [7+8) or (9+10) or 11]

E) 1 + (1.8) 2 + 6 + [(7+8) or (9+10) or 11]

F) 1 + 5 + 6

The heat generation rates due to radiation in the primary concrete were calculated by using a point kernel analysis technique. In addition to the reactor core sources, the code considers the captured gamma and inelastic neutron scattering contributions outside the core, and within the concrete.

A description of the analyses using the STRUDL and GENSHL 5 computer programs and the manual calculations follows:

STRUDL Computer Program

The circular slab was modeled as a gridwork of beams framed in the circumferential and radial directions. The wall and the slab of the reactor pit were also modeled into a space frame connected with the circular mat as a continuous structure. The slab section under the containment wall and the crane wall were modeled to include the stiffening effects of the walls. The roundation mat was supported by vertical and horizontal soil springs, which represent the soil modulus of the elastic subgrade.

The soil spring stiffnesses were varied to achieve a variation of soil pressure distribution to meet the criteria as indicated in Figures 5.2.2-57 (Case II or Case III) and 5.2.2-57A (Case IIA or IIIA). The earthquake evaluation was made considering dynamic soil modulus. Case I "Uniform Pressure Distribution" was not recorded since it resulted in smaller values than either II or III for both static and dynamic conditions.

GENSHL 5 Computer Program

To model the reactor pit, mat, containment wall and dome into the "GENSHL 5" program, which only takes bodies of revolution, the unsymmetrical shape of the reactor pit was replaced by a cylindrical body of revolution. Both translational and rotational soil spring constants were supplied directly to the foundation for Case II or IIA and Case III or IIIA as mentioned under the "STRUDL Program".

Both the "STRUDL" and the "GENSHL 5" programs were run for factored load combinations (a) thru (h), of Section 5.2.2.3, which includes the unsymmetrical tornado and earthquake loads. Earthquake forces were introduced onto the foundation mat as a cosine function. The earthquake evaluation was made considering dynamic soil modulus. The maximum compressive and shear stresses in the concrete and the tensile stresses in the rebars occur in the 10 ft. thick mat at the junction of the mat and the reactor wall for load combinations (b) and (d). According to the "GENSHL 5" program the computed maximum compressive and tensile stresses are -2400 psi and 36,000 psi respectively in the slab cross-section. The maximum vertical shear stress across the 10 ft. mat section is 160 psi. The shear stress in excess of 110 psi is taken by the shear reinforcing.

Manual Calculation (used as a check on the computer analysis)

The irregular shape of the mat, caused by the reactor pit. allows only an approximate method of analysis. The reactor pit area was replaced by an equal sized slab of equivalent stiffness. The foundation mat was then analyzed as a circular plate on an elastic foundation. The reactor pit area was then analyzed separately as a rigid frame.

Liner

The Liner has been designed considering loading due to normal operating, proof-testing and accident conditions. Earthquake or tornado cause straining of the concrete which, because of the anchorage system attaching the liner to the containment wall, is transferred to the liner. The stresses in the liner due to this transfer of strain are ±3550 psi and ±2300 psi for the "Design Basis Earthquake" and the "Operating Basis Earthquake", respectively and are considered in the liner analysis.

All loads were analyzed separately and then combined in accordance with the factored load equations in Section 5.2.2.3.

The liner was also designed, and stiffeners provided as required, to resist the hydrostatic head of freshly poured concrete.

The liner is not considered to participate in resisting lateral shear as a design function when designing the containment wall.

The function of the containment liner is to serve as a leaktight barrier under all postulated operating and accident conditions.

The liner strain capability is limited by the weld material. Although the ultimate strain of the weld material is 17% in 2" (0.17"/"), the limiting allowable strain for this design is conservatively set at 0.5% (0.005"/").

The computed maximum strains in the liner are .003"/" in compression and 0.002"/" in tension.

Stress limits as stated were derived from Table N-424 of the ASME Boiler and Pressure Vessel Code-1968-Section III-Nuclear Vessels.

Commercially available plate varies in thickness; therefore, the buckling analysis for the liner was made considering plate variations of +7 percent and -3 percent from nominal thickness.

The containment reinforcing was designed to yield point stress for the factored load equations. Because a minimum liner thickness of 3/8 inch for the cylinder and dome was imposed by construction consideration, the ratio of liner steel area to reinforcing steel area, for this low design pressure containment concept, is large. This large ratio of liner area to reinforcing steel area, not considered in design, precludes the liner being stressed in tension beyond minimum yield value.

Since the function of the liner is to act as an essentially gas-tight membrane, no credit was taken for the liner's ability to resist primary bursting stresses. This is an extremely conservative assumption since the liner is capable of carrying the design pressure within its tension yield capacity without any assistance from the concrete reinforcing steel. This fact results in two structural systems acting in parallel, either one of which is capable of carrying the design pressure load elastically.

Cycling loads considered in the design of the liner were:

1) Thermal cycling due to annual outdoor temperature variations. Daily variations do not significantly penetrate the concrete shell to influence cycling on the liner. Based on the life of the plant, 40 cycles were considered.

2) Thermal cycling due to containment interior temperature varying during reactor system startup and shutdown, was considered to be 200 cycles.

3) Thermal cycling due to accident condition was considered to be 1 cycle.

4) Cycling due to earthquake was considered to be 10 cycles.

Liner anchorage was designed to accommodate at plate joints, a differential of load due to adjacent plates varying in thickness by 10 percent of the nominal thickness.

Liner stresses around openings were analyzed in accordance with the procedure shown in "Theory of Elasticity" by S. Timoshenko and J. W. Goodier. The analysis neglects the stiffening effect of the penetration sleeve and thus over-estimates the distortion due to the biaxial stress field.

The liner meets selected requirements of the ASME Pressure Vessel Code and, in conformity with the philosophy of this code, the opening was compensated for as required.

The liner plate is anchored to the concrete by additional angles around the penetration.

The bottom liner plate is welded at the joints to continuous structural members which are embedded in and anchored to the concrete base slab.

The juncture of the cylinder and the base slab is fixed. There is no differential translation of the cylinder bottom with reference to the base slab, the rotation which could occur at this juncture is only that rotation which results due to straining of the meridional reinforcing in the cylinder wall at the joint area. The liner juncture at the base was designed to accommodate this rotation.

Figures 5.2.2-59 and 5.2.2-59A and illustrate the liner arrangement used under the reactor and at the base cylinder line juncture. The behavior of the liner arrangement at the base of the containment wall was investigated for accident conditions. The cylindrical knuckle which serves as a transitional member between the wall liner and the mat liner was considered as an arch with fixed supports. The stresses in the knuckle are tensile stresses in the accident case therefore the danger of buckling is averted. Local cracking of the concrete at the anchors would not result in loss of the anchor because of the length of the anchor and because the anchors are tied back into a greater depth of the concrete wall, by means of Nelson Studs welded to the anchors. The arrangement at the bottom of the reactor pit is somewhat different. The wall liner meets the liner of the floor slab at right angles. Both liners are welded at their junction at an anchorage angle embedded in concrete. Since the liner is protected from accident temperature by the concrete fill, the only stresses that exist are the axial stresses which are induced in the liner plate by normal temperature gradients. These direct compression stresses induced by the restraining concrete are low enough to be carried by the steel plate without either buckling or yielding of the liner between anchorages. The strain limits developed for this design are, as previously mentioned, conservatively set at 0.5% strain.

Lateral load transfer under the interior structure is accomplished by a series of interrupted keys in the base slab. The load transfer is therefore by direct bearing. The maximum bearing stress is 2200 psi. The liner follows these keys so that there is no loss of liner leaktight integrity. See Fig. 5.2.2-59B.

The fill concrete in the core of the containment is locked between the crane wall and the primary shield and therefore transfers its lateral load to both the crane wall and the primary shield.

The lateral load of the fill concrete in the annulus between the crane wall and the containment wall is transmitted to the crane wall by rebars embedded into the crane wall. See Figure 5.2.2-59B, attached.

Uplift forces are not transmitted through the liner plate. All equipment uplift forces are transmitted by means of weldments anchored directly into the concrete.

See reactor coolant pump and steam generator support anchorages, Figure 5.2.2-59C , and ice condenser support column anchorage, Figure 5.2.2-59D.

The crane wall experiences some net uplift force and is therefore anchored to the foundation slab by dowels which are welded to but do not penetrate the liner. See Figure 5.2.2-59E attached. The maximum computed stress in the dowels is 10,000 psi.

The seams of the bottom liner plate are covered by weld channels. To prevent imposition of lateral loads onto this channel due to thermal expansion of the bottom fill concrete over the liner, or earthquake, the weld channels were encased, where required, in styrofoam material before placing of this fill concrete.

Where loadings must be transferred through the liner, they are transferred through the liner in a direct path by means of structural weldments embedded into the concrete. The leaktight integrity of the liner is not impaired.

Internal Structure

In addition to the three basic containment vessel structural components, there exists an internal structural system consisting of the reactor shield, divider barrier and other internal components. This internal system is completely separated from the containment vessel shell at all elevations above the base slab, so as to prevent restraints or concentrated loads from being imposed on the containment vessel cylinder wall. The internal structure is a self-supporting reinforced concrete structure capable of withstanding all loads to which it is subjected. The dynamic analysis considered independent movement of interior and exterior structures and the maximum values of deflection so determined were used to determine the required separation of the two structures ("RATTLE SPACE").

There exists an annulus space of 13 ft. between the crane wall and the containment wall. Within this annulus space are two slabs and a number of radial walls all framing to the crane wall, but all maintaining a nominal 4-inch gap to the liner. When allowance is made for construction tolerances and liner weld test channel depths a clear rattle space of at least 1-3/8" remains.

Static and thermal loading conditions for the crane wall were analyzed in accordance with the procedure in "Theory of Plates and Shells", second edition by Woinowsky-Krieger. The crane wall was considered to be a complete cylinder for initial analysis. The section at the equipment hatch area was then removed and a vertical section of the cylinder at each side of this opening was considered to act as a vertical beam spanning between the crane girder and the floor. Restraint to radial deformation at the edges of the opening is provided by the end closure walls of the ice condenser compartment which are oriented radially and considered to cantilever from the operating deck. Discontinuity moments were considered at the edges of this large opening and at the steam generator and pressurizer enclosures.

Values of forces and moments determined from manual computations for static and thermal loading conditions were used as a check for those values determined from a computer analysis. For the computer analysis the internal structure was modeled as a space frame composed of a network of prismatic members. The computer program used was the "American Electric Power General Frame Analysis."

The internal cylinder was modeled as a grid work consisting of horizontal beams at intervals along the height and vertical beams intersecting the horizontal beams and extending from the top of the cylinder to the base slab or terminating at openings.

The steam generator and pressurizer enclosures were similarly modeled. The horizontal members were framed to the interior cylinder at the nodal points (intersections between vertical and horizontal members). The vertical members of the enclosures were framed to the nodal points of the floor (barrier slab) grid.

The floor slab and reactor primary shield were modeled in a similar manner.

The forces, moments and shears determined from the seismic dynamic analysis were superimposed on those determined from analyses of other loads in accordance with the factored load equations as indicated in Section 5.2.2.3.

A flexible barrier between the bottom of the ice condenser compartment and the containment cylinder wall is provided, to prevent the flow of steam and air from bypassing the ice condenser. The flexible barrier was so designed that no load transfer occurs across it. See Figs, 5.2.2-60 & 5.2.2-60A. The extent of the seal is shown in Figs. 5.2.2-60B & 5.2.2-60C.

The seal does not carry pressure but is backed up by a steel plate with which it is in contact. The seal assembly was designed to withstand a peak pressure of 24 psi. The seal material is expected to have a minimum life under operating conditions in excess of 10 years. The seal material is Uniroyal #3807 or equal.

Under operating conditions the seal sees very little radiation, however, there may be some areas which would be exposed to a dosage of 40 MR/HR (0.0014×10^7 Rad. for 40 years continuous plant operation). Under accident conditions based on TID-14844 activity release assumptions the seal material will see a dosage of 1×10^7 Rads. in 10,000 seconds. The criteria for the seal material is that it remains functional during operating conditions within its material life and that during accident conditions the seal material remains functional for two hours. It is required that the bypass of the ice condenser be limited for the period of ice melt-down which is 5,500 secs. (approx. 1 1/2 hrs.).

Some properties of Uniroyal #3807 are listed below:

1. <u>Tensile</u> 200 lbs/sq. in.

2. <u>Elongation</u> 205%

3. <u>Tensile at 100% elongation</u> 139 lbs/sq. inch

4. <u>Aging Characteristics at 120°F</u> UNIROYAL 3807 is based on ethylene - propylene terpolymer. These polymers are noted for their outstanding aging characteristics. Test data indicates that the EPT, on which 3807 is based, is good for service in excess of 10,000 hours at 120°F. It will retain 100% of its elongation after continuous service for 10 days at 350°F and for 100 days at 300°F.

5. The following gives an indication of the basic properties for resistance to moisture in the presence of dilute borate solutions. UNIROYAL 3807 shows virtually no change in properties after the following cycle: 2.5 hrs at 286°F, followed by 24 hours at 212-220°F, followed by 40 hours at 145-158°F. The test sample material was immersed during the course of this test in a solution of 1.43% boric acid at a pH of 9.3. This pH was obtained through the use of 3.8 grams per liter of sodium hydroxide.

   The exposure at 221°F in radiation environments with continuous dosage amounts of 1×10^8 rads relates to a specific test and does not indicate a limitation of temperature to which UNIROYAL 3807 may be used, as indicated above, continuously at 120°F and intermittently at 350°F.

   As regards fire resistance and ultimate temperature limitations of the materials, results of tests conducted in Uniroyal Laboratories on 3807, utilizing the DOT definitions and tests for oxygen index, smoke temperature, melt temperature, and ignition temperature, are as follows:

Oxygen index	18.7
Smoke temperature	503°F
Melt temperature	529°F
Ignition temperature	620°F

The spacing between bolts holding the seal is 3". The maximum lateral movement between the containment wall and the crane wall is 1.3". This represents less than 44% elongation in 3". The material is good for 205% elongation. Tension in the material during elongation would be 60 psi if it were laid in without additional length between bolts. The material capability is 200 psi.

The seal, however, is laid in with 1/2" play (extra length) between bolt dimensions. Therefore the seal sees very little actual cyclic elongation.

The seal is completely accessible for inspection and replacement. The divider seal is inspected at least once every eighteen months, during a unit shutdown.

If the seal material were not provided the hypothetical by-pass area would be 32 $ft.^2$

It is assumed in the post LOCA containment pressure analysis that there will be no steam bypass of the structural seals provided between the crane wall and containment wall.

Dynamic Analysis for Seismic Loading

Computer runs were made with various soil shear moduli. The analysis showed that for a variation of ± 10% of the soil shear modulus the structure natural frequency varies within a range of ± 5%. Assuming also that the concrete modulus of elasticity varies within ± 10% of the recommended value, this results in an additional ± 5% variation in the natural frequency of the structure. Based on the above results it was concluded that the natural frequency of the structure must have a tolerance of ± 7.5% to reflect the possible variations in the soil and concrete properties.

Containment Structure

The vessel was analyzed to determine the structural response to earthquake loading. A multi degree-of-freedom model of the structure was used. The interrelation of the containment vessel structure and the interior structure through the base was considered, as was the rotation and translation of the composite structure on the subgrade.

The containment structure was modeled as two cantilever beams coupled at the base by a rigid foundation mat. A modal analysis was made using response spectra to determine the maximum probable peak accelerations at various elevations of the structure by means of an AEP computer program "Containment Vessel Program". 4% modal damping was used for all modes for the "Operating Basis earthquake" (10% G) coincident with LOCA and 7% modal damping for the "Design Basis earthquake" (20%G) coincident with LOCA. Computed forces from the accelerations so determined were used as input to a shell of revolution model of the structure to determine the stresses.

The dynamic model is shown in Figure 5.2.2-61.

An evaluation was made of the natural frequency and mode shapes of the first three modes. These frequencies were used in conjunction with response spectra and the appropriate damping factor to evaluate maximum displacements, velocities, and accelerations. The values of these parameters determined for each of the first three modes, were adjusted by the modal participation factor and mode shape to obtain the moment and shear in each mode. The moments and shears of the individual modes were combined by computing the square root of the sum of the squares of the individual modal values as indicated in a paper by Dr. Nathan M. Newmark (25 May, 1967) "Design Criteria for Nuclear Reactors Subjected to Earthquake Hazards". The effect of the higher modes were evaluated at this point by examining their contribution.

The formulation of the natural frequency equations makes use of the stiffness method of analysis. The soil spring constants used for the rotation and translation of the structure were based on the results of field investigation. The percent of critical damping factor used for the 10 percent and 20 percent of gravity seismic conditions are a maximum of 4 percent and 7 percent respectively. The earthquake ground response spectra for this site are shown in Chapter 2.

Possible coupling of the internal structure and the containment vessel structure through the ice condenser internal support structure was considered.

The spring constant representative of the material used for the thick layers of insulation in the ice condenser compartment is 6 psi per inch of deflection. For a spring constant of this magnitude, it has been determined that the effects on the natural frequencies are less than 0.003 percent for the first mode, less than 0.60 percent for the second mode and less than 3.5 percent for the third mode. Moments vary by less than 1.5 percent and the shears by less than 0.20 percent.

The effects, therefore, were considered to be negligible and the mathematical model for seismic analysis considers the interior structure and the containment vessel structure to be uncoupled at all elevations above the base slab.

Torsional effects of unsymmetrically located items although of too small a mass to effect the containment structure significantly were analyzed for the local structural elements.

As stated in the discussion of the general analytical model, (Appendix "F" of the Original FSAR), to evaluate the effects of cracking of the concrete, provisions were made in the seismic program to input various percentages of concrete area for the structure.

Structural deflections, due to shear and flexural deformations, were determined for the containment vessel structure and for the interior structure at incremental intervals along the height. The deflections were determined for the individual modes and for the composite response. In the composite response the rotational offset and the translational offset were included.

It was considered that under the design basis earthquake condition, the reinforcing may be stressed to yield point values and that under operating basis earthquake the reinforcing may be stressed somewhat below the yield point value. Since the maximum damping values stated include the effects of the soil, they are considered to be conservative.

The percentage of critical damping for use in the seismic analysis of the reinforced concrete structures is dependent upon the stress in the reinforcing.

The percentages indicated in Table 5.2-4 were used for the design analysis of the structures. For the condition of approximately yield stress level in reinforcing, the maximum value of percentage critical damping is 7 percent. For stress levels of approximately 1/2 yield stress level, the value of percentage critical damping is 4 percent.

Two sets of percentage critical damping are indicated for the containment structure because, under a condition of seismic occurrence coincident with accident condition, the amount of cracking in the structure is much greater than for the condition of seismic occurrence without a coincident accident.

The D. C. Cook auxiliary building is a complex structural system, asymmetric in plan, with heavy concrete slabs at various floor elevations. These floor slabs are interconnected with numerous concrete shear walls and or heavy cross-braced steel members. The overall height dimension is smaller than the plan dimensions. This low height to plan aspect ratio indicates that under lateral loads the predominate deformations of the long shear walls will be shear deforformation. Consequently, the relative rotations of the slabs about horizontal axes do not cause significant deformations, but because of the asymmetrical mass-stiffness distribution, rotation of the slabs about a vertical axis could occur when this type of structure is subjected to lateral loads. Therefore, if a shear structure is modeled in an X-Y-Z axis system where the Z axis is vertical and the X and Y are parallel to the principal axes of the structure, three degrees of freedom, rotations about the X and Y axes, Θ_x and Θ_y, and vertical translation, Δ_z, could be neglected in the model. The motions of the lumped masses in the model are restricted to a horizontal plane and each lumped mass is allowed the remaining three degrees of freedom Δ_x, Δ_y and Θ_z.

In discussing the Cook auxiliary building model the words "Model Slab" will be substituted for the words "lumped mass" because the mass of the actual structure is simulated in the model with virtual infinitely rigid slabs located at the elevations of the major floor slabs and roofs of the structure. The actual slabs are considered to be infinitely rigid in their own planes. The rigid body motions of the model slabs consist of three degrees-of-freedom; horizontal translation in two perpendicular directions and rotation about a vertical axis. The model slabs are interconnected by weightless elastic springs which possess stiffness in the X or Y direction and simulate the shear walls and vertical bracing in the structure. These springs are distributed horizontally on the model slabs so the torsional stiffness interconnecting two slabs is approximated.

Since the ends of the springs are considered to be horizontally distributed on the special extent of the model slabs, the model slabs are not point masses, but may be thought of as rigid bodies with horizontal dimensions where a vertical dimension is meaningless because the mass of the actual structure is considered lumped in the planes of the model slabs.

Mass Properties:

Three coordinates are required to describe the motion of each model slab. Therefore, three mass parameters are associated with each model slab. These mass parameters for the ith slab of the model are:

M_{x_i} associated with X translation

M_{Y_i} associated with Y translation

$I_{\Theta i}$ associated with rotation about a vertical axis

The mass parameter associated with X translation and Y translation is the same and equal to the mass of the slab. The mass polar moment of inertia, I_o, is about a vertical axis through the centeroid of the slab.

Stiffness Properties:

When the stiffness of the structural components which interconnect slabs were evaluated, the following assumptions were made:

1. All floor and roof slabs were rigid in their own planes; no joint can displace relative to another joint on the same slab.

2. Walls interconnecting slabs offer resistance to relative displacement of slabs in the direction of the long dimension of the wall only.

3. The stiffness of small reinforced concrete columns or walls and steel can be neglected because their stiffness is small compared to the stiffness of larger walls.

When resisting lateral loads applied parallel to the long dimension, most walls act as short, deep beams; therefore, the contribution of shear to the deflection must be considered in calculating the stiffness of a wall. The stiffness of an individual wall was calculated by the following formula:

$$K = \frac{1}{\Delta}$$

Where $\Delta = \frac{Fh}{GA}$

and F = shear form factor
A = cross-sectional area of the wall
G = shear modulus of concrete
h = height of wall

The stiffness of steel framing which acts as springs is evaluated with frame or truss analysis computer programs.

The horizontal flexibility of the soil which supports the auxiliary building was simulated with linear elastic springs distributed over the base of the model in two perpendicular directions.

Analytical Procedure:

The compilation of the mass-stiffness properties of the auxiliary building began early in the design process. As design proceeded, the dynamic model was kept current with design changes. To facilitate the calculation, documentation and revision of the model's properties throughout the design process, a computer routine was used in compiling input to the dynamic model.

Mass properties of each slab are coded on a card and the program uses this data in compiling the mass matrix and the load vector.

Each structural component which is considered a spring in the model is assigned an identification number. For each spring the identification number, stiffness, slabs the spring interconnects, and the horizontal distances of the springs end from the slab centroids (required to formulate torsional stiffness about a vertical axis) are coded on a card. The program uses this information to compile the stiffness matrix, and after the dynamic response is calculated, again uses this information in distributing the inertia forces into the structural components by imposing calculated modal displacements on the springs.

Input data to the dynamic program consists of 10 groups of cards which specify the mass-stiffness properties of the model, degrees of freedom, and the loading. This input is titled and printed as output in the following order:

1. Problem identification
2. Number of X springs

3. Number of Y springs
4. Rocking code
5. X spring constants and topology
6. Y spring constants and topology
7. Seismic loading code (direction of load)
8. Structural symmetry code
9. Slab masses and polar moments of inertia
10. Number of modes to be considered
11. Number of spectral data points or time history data points
12. Spectrum data or time history forcing functions
13. Slabs where responses are required.

Output from calculation done by the dynamic program consists of the following groups of information for each direction of excitation (X or Y or both):

1. Stiffness matrix (optional)
2. Mass matrix (optional)
3. Loading vector (optional)
4. Orthogonality check of eigenvector
5. Modal periods
6. Modal participation factors
7. Mode shapes normalized with respect to the mass matrix
8. Modal displacements
9. Modal inertia forces acting on the masses
10. Probable maximum displacements and inertia forces at slab centroids
11. Probable maximum shear forces in springs
12. Time history response if time history forcing function used as excitation
13. Slab response spectra (optional)

Seismic forces used in the structural design of the auxiliary building were obtained from exciting the dynamic model with the Operating Basis and Design Basis Earthquake Spectra presented in the PSAR. Two percent of critical damping was used in the analysis.

The equipment design criteria for Class I systems and components supported in the auxiliary building were developed by generating response spectra from the motions of the lumped masses in the dynamic model. Mass motions used to generate response spectra were obtained from a time history analysis of the dynamic model.

Input data for the seismic evaluation of Class I equipment was derived from the computer program. The information for equipment seismic input are natural frequency for each of the first three modes and response curves for the required elevation for the required equipment-damping values.

5.2.4 PENETRATIONS

In general, a penetration consists of a sleeve embedded in, and anchored to, the concrete containment wall, and welded to the containment liner. The weld to the liner is shrouded by a channel which can be pressurized to demonstrate the integrity of the penetration to liner weld. The core pipe, electrical conductor cartridges, or air ducts pass through the embedded sleeves. The ends of the resulting annuli are closed off by welded end sections. Provision was made for differential expansion and misalignment between pipe, cartridge, or duct and sleeve. No significant loads are imposed on the liner. Pressurizing connections have been provided to periodically demonstrate the integrity of the penetration assemblies.

An elastic stress analysis was performed for each penetration assembly using a finite element computer program. The design basis accident conditions used in this analysis consider moment, shear, axial thrust, and torsion resulting from individual breaks either inside or outside the containment. Normal and tangential stresses in the penetration assembly, as well as concrete bearing stresses, were determined for the following three emergency loading cases (as indicated in Figure 5.2.2-62):

1. Moment + Shear

2. Moment + Axial Thrust

3. Moment ± Shear ± Torsion

In determining penetration stresses, no consideration was given to the ability of a thickened liner to aid in resisting the applied loading.

The allowable stress intensities for the materials used in the penetration assemblies were determined from the criteria presented in ASME Pressure Vessel Code, Section III 1968 Ed, Figure N-414, Table N-421, and Table N-424, and the allowable stresses of USAS Piping Code B 31.1-1967 Ed.

Case	Core Pipe	Sleeve and Flued head
	B 31.1	ASME III
Normal	$S_{allowable} = 1.0\ S_{tabulated}$	$P_m = S_m$ $P_L = 1.5\ S_m$ $P_L + P_B = 1.5\ S_m$ $P_L + P_B + F = S_a$ $P_L + P_B + Q = 3.0\ S_m$
Upset*	$S_{allowable} = 1.2\ S_{tabulated}$	Same as above
	ASME III	ASME III
Emergency*	$P_m = S_y$ or $1.2\ S_m$ $P_L = 1.5\ S_y$ or $1.8\ S_m$ $P_L + P_B = 1.5\ S_y$ or $1.8\ S_m$	Same as core pipe.

*Includes seismic effects

Figure 5.2.2-62 illustrates the design details for typical penetrations.

For example, for penetration CPN-48 (Residual Heat Removal) the folfowing maximum local stresses were computed for the pipe rupture condition.

		Maximum Stress Intensity (ksi)	Allowable Stress Intensity (ksi)
Flued Head:	Core part	29.7	43.7
	Sleeve part	25.8	52.5
Core Pipe		42.8	43.7
Exterior Support Plate		6.7	52.5

Thermal protection of concrete at hot penetrations is provided by means of two redundant cooling coils. Each individual coil is capable of maintaining adjacent concrete temperature to a maximum of 150°F. Therefore, in the unlikely event of a failure of one of the coils, the faulty coil can be isolated without loss of thermal protection to the concrete.

The thermal gradients of each hot penetration, for its operating condition, were determined to establish the cooling capacity required to maintain concrete temperatures at less than 150°F, assuming a 120°F ambient condition.

Stress analyses using the "GENSHL 5" computer program were performed to determine the stresses and strains in the penetration sleeves for the various factored operating and accident loading conditions. At the junction of the thickened liner and penetration sleeve, the strains determined were a maximum for the accident loading condition. The worst case occurred at an electrical penetration sleeve with a strain of 0.107%. The worst strain for a piping penetration sleeve was 0.055%. The allowable strain has been set at 0.5%.

Stresses in the plastic domain were not combined since the analysis performed did not require consideration of the full plastic strength of the pipe.

Normal shear, bending and torsional reactions from the pipe ruptures are transferred from the pipe sleeve to the containment wall by the system of circumferential and longitudinal lugs on the penetration sleeve.

NORMAL LOADS are transferred to the concrete by bearing under rings attached to the sleeve, the concrete at the perimeter of the ring is checked for punching shear and diagonal tension. When, for accident loads, the punching shear is greater than 500 psi or the diagonal tension is greater than 60 $\#/in^2$, shear reinforcing has been added. The normal load imposes local bending on the wall, the magnitude of the resulting stresses were analyzed by elastic beam formulae and, where necessary, extra rebars were added in both the hoop and meridional direction.

SHEAR AND BENDING PIPE LOADS are transferred to the wall by a combination of bearing under the sleeve and radial shear at the perimeter of the rings in the concrete. The same criteria is used as outlined for "normal forces" above for transferring these stresses to the reinforcing. The allowable bearing stress = $0.9 \times 0.85 f'_c = 2680$ $\#/in^2$.

TORSIONAL PIPE LOADS are transferred from the sleeve to the concrete by bearing under the longitudinal stiffeners. The allowable bearing is the same as above. These bearing stresses then induce shear and tension stresses in the wall, but in all cases these stresses were found to be very small and no additional reinforcing was required.

Where reinforcing bars were bent to clear penetrations the radial compressive stresses in the concrete, under a bar that has been stressed to the yield point, has been limited to 2500 psi.

The minimum radius of curvature is 3'-0".

In addition to the large bend radius, curved bars have been tied back to adjacent straight bars using #6 ties. See Figure 5.2.2-63.

Where penetrations larger than 4'-0" in diameter are required, the following loadings were considered in the design of the openings: Pressure, dead load, operating basis earthquake, design basis earthquake, wind, tornado operating temperature, accident temperature and shrinkage. The secondary forces were treated by computer program as part of its analysis.

The thickened liner around the penetrations was proportioned in accordance with the area replacement method given in the ASME Pressure Vessel Code, Section VIII. For the mechanical and electrical penetrations a stress analysis using the "GENSHL 5" computer program was performed to determine the stresses in the thickened liner and the penetration sleeve, resulting from the various factored operating and accident loading conditions. The thickened liner around the Equipment Hatch and Personnel Lock was modeled with the thickened concrete shell, and a finite element analysis was performed for the composite section using the FELAP computer program of the Franklin Institute. Liner stresses were computed for the factored operating and accident loading conditions. The stresses for the thickened liner and sleeve materials were compared with the stresses given in Table N-421, and Table N-424 of the ASME Pressure Vessel Code, Section III - 1968. In all cases the stresses obtained from the stress analysis were less than those specified. As a check on the computer analysis, approximate hand calculations were performed for the operating thermal loads considering the thickened liner as a flat plate with the edges restrained and a uniform temperature change across the thickness. The following equation was used for these approximate calculations:*

$$f_1 = f_2 = \frac{E \Delta T}{(1- \nu)}$$

Where:

f_1 = Principal meridional stress

f_2 = Principal Hoop Stress

α = Coefficient of Expansion

E = Modulus of Elasticity

ΔT = Uniform Temperature change across the plate thickness.

ν = Poissons Ratio

Tables 5.2-8 and 5.2-9 summarize the membrane stresses in the thickened liner for typical penetrations. In those areas where the yield stress has been reached the resulting strains were checked and are less than 0.5%.

The computer program used to calculate the liner stresses assumes that there is a compatibility of strain between the liner and the concrete wall. The liner is mechanically attached to the concrete wall by anchors and therefore is less stressed than the computed values.

To determine the critical buckling stress between anchors, the liner was analyzed as a flat plate. This assumption is conservative in that the liner will have to buckle against its own curvature. For the analysis it was assumed that the liner was fixed at the angles and there was no differential radial movement of the boundaries. The analysis was based on an interaction curve given by A. Pfluger "Stabilitats probleme der Elastostalik", pages 404 and 405, Springer

* Timoshenko and Goodier, Theory of Elasticity, Second Edition, McGraw-Hill Book Co., p. 401.

Verlag Berlin 1964. The critical stress resultants N_1 and N_2 are the stresses induced in the plate (see Fig 5.2.2-64) and are defined as $N_1 = K_5 N_e$ where $K_5 = 6.97$.

$N_2 = K_3 N_e$ where $K_3 = 4.00$

$$N_e = \frac{\pi^2 Et^3}{12\ (1-\nu^2)} \times 1/b^2$$

Where: E = Modulus of Elasticity

ν = Poissons ratio

t = Plate thickness

b = Plate Width

a = Plate length

It can be seen from the interaction curve that for a = infinity the influence from N_1 can be neglected.

N_2 (Critical) = 60,000 psi

b = Span = 14" = spacing between anchors

(See Fig. 5.2.2-64)

The stress in the liner at operating temperature is -18.5 ksi ... the factor of safety against elastic buckling = $\frac{60,000}{18,500}$ = 3.24.

The specified design stress limits are ± 20 ksi for operating condition and yield stress for accident condition.

The thickened liner between the penetration and the transition to 3/8" thickness is anchored by inverted angles with the leg welded to the liner and spaced at 14".

The unbalanced shear forces at the transition from the thickened liner to the typical 3/8" wall liner thickness are taken by Nelson Studs.

The maximum shear force on a panel occurs during accident conditions when one panel is completely buckled while the adjacent ones remain unbuckled. The unbalanced shear force is transmitted to the concrete by bearing between the angle and the concrete.

The material for both the thickened and unthickened liner plate is A442 Gr. 60 and the material for the penetration sleeve is A333 Gr. 6 or A516 Gr. 70.

The stresses in the reinforced plate are transferred to the concrete wall by the angles and Nelson Stud Anchors, described above, and to the typical 3/8" wall liner through the butt weld connecting the two plates.

The maximum strain is 0.11% for an unbuckled panel and 0.3% at the plastic hinges in a buckled panel. The allowable strain is 0.5%.

A. The Franklin Institute finite element computer program was used to analyze all stresses in the rebar and concrete around the equipment and personnel accesses. The procedure used was to analyze, by the FELAP program, rectangular areas of the wall 75' (Horiz) x 64' (Vert) and 54' (Horiz) x 46' (Vert) for the equipment hatch and the personnel hatch, respectively. These areas were then divided into elements approximately 4' x 2'6" in elevation. Different material types across the wall were represented as separate layers. Boundary conditions, taken from the GENSHL program results, material properties, loads and temperatures were input for each load condition; from the results, concrete layers carrying tension were cracked and the rebar modified until the stresses were within the allowable.

The openings were checked for operating loads, accident loads and test pressure loads.

1) <u>At Normal Operating Condition</u>

The loads due to normal operating conditions are:

(a) Operating temperature
(b) Dead load
(c) Shrinkage
(d) Creep

The allowable stresses in the reinforcing steel and concrete due to the worst combination of operating loads were 0.5fy = 20,000 psi for steel in tension and 0.45f'c = 0.45x3500 = 1580 psi in concrete in compression.

2) <u>Test Pressure</u>

The thickened concrete around the openings was analyzed for the following loads under test condtions:

(a) Internal pressure of 1.34 times accident pressure equal 1.34x12 = 16 psi
(b) Dead load
(c) Live load
(d) Temperature transients at test conditions
(e) Shrinkage

The allowable stresses due to the combinations of the above loads were increased 33% above the operating stresses since the test pressure is temporary.

3) At Factored Loads

The factored load combinations for ultimate design are as follows:

(a) $1.5P + DL \pm 0.05DL + (T' + TL')$

(b) $1.25P + DL \pm 0.05DL + (T'' + TL'') + 1.25E$

(c) $1.0P + DL \pm 0.05DL + (T''' + TL''') + E'$

The thickened concrete around the openings was checked for the above load combinations.

The capacity reduction factors used in the ultimate design were 0.95 for axial stresses, 0.9 for bending stresses, and 0.85 for diagonal tension, giving allowable stresses in the rebar of 38,000 psi, 36,000 psi, and 34,000 psi, respectively; and 0.9x0.85x3500 = 2680 psi compression in the concrete.

PERSONNEL HATCH

The computed maximum meridional and hoop stresses in the rebar were 34,000 psi and 37,200 psi, respectively (load Combination "a").

EQUIPMENT HATCH

The computed meridional and hoop stresses in the rebar were 26,000 psi and 38,000 psi, respectively (load Combination a).

The initial run of the FELAP Computer Program was with uncracked concrete section and manually estimated reinforcing; the results showed which layers carried tension. These were then cracked in both hoop and meridional directions for the next run. In subsequent runs reinforcing and cracking were modified until the stresses were within acceptable limits. Therefore, it can be seen that the concrete is conservatively assumed not to carry biaxial or uniaxial tension, but that these stresses are carried by the reinforcing under design criteria.

The FELAP Computer Program combined all normal and shear stresses due to axial load, two directional bending, two directional shear and tension. In places where the shear stresses were greater than 60 psi, #8 sloping radial bars were added; these were required only under the personnel hatch. Also, extra diagonal bars were added where the tangential shear stress was greater than 40 psi. The design criteria for the thickened concrete around large openings was the same as for the rest of the containment wall.

The FELAP Computer Program was used to design the thickened part of the containment wall around the openings.

This was checked by comparing stresses at similar points on the GENSHL AND FELAP Programs.

The thickened portion of the wall had little effect on the typical wall rebar stresses, except on the vertical sides of the equipment hatch where additional rebar was required to keep the rebar stresses below yield.

The effect of shrinkage is to impose tensile stresses in the concrete and compressive stresses in the rebar and liner. Since the compressive stress in the rebar reduces the tensile stresses due to accident loads they were neglected, but, the tensile stresses in the concrete will reduce the margin against cracking when the accident loads are imposed on the structure. The tensile stresses in the concrete due to shrinkage were calculated from the following formula:

$$f_c = \frac{\eta \, S_s \, \tau_s \, E_c}{(1-\nu_s)A_c + (1-\nu_c)\eta \, A_s} = 80\#/\text{sq.in.}$$

f_c = stress in concrete

η = ratio of modulus of elasticity of steel and concrete

A_s = total cross sectional area of steel

τ_s = shrinkage strain in concrete

The value of $\tau_s = 10^{-4}$ was taken from the paper by "Matlock and Hansen"* which states that for a given water/cement ratio and aggregate, shrinkage decreases linearly as volume/surface ratio increases. The volume/surface ratio for the containment building wall at the personnel access is 72.

The maximum volume/surface on the graph equals 8, to be on the conservative side, a volume/surface = 24 was used, which, by extrapolation, gave a shrinkage strain of 10^{-4}.

Torsional stresses were evaluated by the multi-layer FELAP Computer Program. The analysis this program performed was too complex to check by approximate manual calculation, such as comparing the thickened concrete to a circular plate.

Details of the reinforcing pattern used around large openings such as the Personnel and Equipment Hatches are shown in Fig. 5.2.2-65 & 5.2.2-65A.

A Factor of Safety of 1.5 has been applied to the accident pressure when combined with the associated accident temperature in factored load Combination (a); a Factor of Safety of 1.25 when combined with associated accident temperature and operating basis earthquake in Combination (b); and a Factor of Safety of 1.0 when combined with associated accident temperature and design basis earthquake in

*" Shape of Member on the Shrinkage and Creep of Concrete', By Hansen & Matlock, ACI Journal 63/10 Feb. 1966.

Combination (c). The earthquake stresses around the openings are negligible, therefore Combination (a) controls and 1.5 is the minimum Factor of Safety.

The allowable stresses for these combinations are ϕf_y or $\phi f'_c$ where ϕ = 0.95, 0.9, and 0.85 for axial, bending and diagonal tension, respectively.

The maximum stress taken from the FELAP Computer output equals 38,000 psi for load Combination (a) and, therefore the stress in the rebar at design load would equal $\frac{38,000}{1.5}$ = 25,300 psi giving a Factor of Safety against minimum yield = $\frac{40,000}{25,300}$ = 1.60.

Equilibrium checks of internal stresses and external loads were made both for the "GENSHL" program and the "FELAP" program. All bodies as modeled in the "GENSHL" program were checked for compatibility. This particular check was necessary for determining whether the lengths of the bodies selected for the structure modeling were satisfactory. Additional and more detailed checks of the "GENSHL" Program were made at the following spots:

1) Locations of discontinuities in the geometrical shape such as, the shell wall and base mat juncture, and the shell wall and dome juncture.

2) Locations of major change in temperature conditions. In the meridional direction this occurs at the lower and upper limits of the ice condenser area (El. 642'-0" and the springline). In the hoop direction this occurs near the limits of the ice condenser area (Azimuths 150°F and 210°).

3) Locations of localized accident conditions. The fan accumulator room where, the unsymmetrical pressure and thermal loading cause high stresses. (El. 630'-0" and Azimuth 90).

For the FELAP Program complete checks were made at the following locations:

1) At the Personnel Hatch.
El. 618'-0" at Azimuth 325°. This is in the thickened portion close to the haunch where stresses are relatively high.

2) At the Equipment Hatch.
El 644'-0" at Azimuth 145° which is the juncture of the containment shell and the haunch for the thickened portion at the Hatch and also where the effect of the thermal condition in the ice condenser compartment is felt.

PROCEDURES FOR CHECKING RESULTS OF "GENSHL" PROGRAM

A) Compatiblity (Internal Check)

1) Moments and forces acting on the end of any element of the shell, and its deformations, are exactly equal to those at the adjacent end of the next element, as listed in the results of the Computer analysis.

2) The sum of the products of the internal stresses of all the cross-sectional layers of any element times the corresponding layer thicknesses is equal to the force resultants, axial or shear, given by the computer analysis.

The sum of the products of those internal forces of all the layers times the corresponding arms from the centroid of the section is equal to the moment acting in that direction given by the computer.

3) The sum of all the force resultants due to each individual load times the factors for the specific combinations is equal to the final results from the superposition of all individual loads by the computer.

B) Equilibrium (External Check)

The summations of moments and forces given by the computer acting at any element of the shell as a free body are statically in equilibrium with external loads.

1) The axial force in the cylindrical shell, per ft. of circumference, due to internal pressure, given by the computer, is equal to 1/2 PR (meridional direction), and PR, per ft. of height of shell, (hoop direction).

P = internal pressure (psi).
R = internal radius in inches.

They are equal to zero for uniform thermal loadings.

For non-uniform or unsymmetrical thermal loadings the sum of the membrane forces throughout the whole cylindrical section of each harmonic function from the computer analysis is equal to zero.

2) In the check of seismic computations the containment shell is divided into thirteen segments. The sum of the weight of each segment times the acceleration times the arm from the center of gravity of each segment to the base mat is equal to the sum of the resultant moments given by the computer analysis in the meridional direction.

The sum of the weight of each segment times the acceleration (% G) is equal to the sum of the resultant tangential shears at the base mat given by the computer analysis.

3) The curves plotted, based on the foundation settlements from the computer analysis times the corresponding soil modulus of elasticity, are close to the shapes of foundation pressure distribution stated in the FSAR.

The sum of those computed soil pressures times the corresponding foundation areas is equal to the total loads acting on the mat plus the weight of the foundation mat itself. The discrepancies between the manually computed values and the GENSHL results are less than 10%. The greatest discrepancies appear at points of differences, either in the discontinuity of geometric shape, in the varying stiffness of different layer properties of adjacent elements, or in varying loading conditions between adjacent elements.

The computer results take all these into consideration, make all the necessary compatibility corrections, and add the local bending effects of the shell in addition to the membrane forces.

PROCEDURES FOR CHECKING RESULTS OF "FELAP" PROGRAM

A) Internal Check

1) The sum of the internal stresses of all the layers given by the computer times their corresponding layer thicknesses, is equal to the force resultant in the same direction given by the computer at the middle of panel.

2) Take any panel or a group of panels as a free body. The sum of all the forces given by the computer acting at the four nodal points of a panel is statically in equilibrium. Likewise, the sum of all the forces given by the computer at the exterior nodal points along the boundaries of a group of panels is statically in equilibrium.

3) Pass a horizontal section through the middle of panels within a certain area. The sum of stress resultants in the meridional direction given by the computer times the width of the corresponding panels is in equilibrium with the sum of forces in the meridional direction acting at exterior nodal points along the boundaries of that sectioned area.

4) Pass a vertical section through the middle of panels within a certain area. The sum of stress resultants in the hoop direction given by the computer times the height of the corresponding panels is in equilibrium with the sum of forces in the hoop direction acting at the exterior nodal points along the boundaries of that sectioned area.

5) The sum of the stress resultants of each individual load times the factors for the specific combination is equal to the final results from the superposition program given by the computer.

B) External Check

The sums of the forces given by the computer acting at any nodal point joining any four panels are statically in equilibrium with external loads. It is equal to zero in the uniform thermal loading and nodal point or the external loadings acting on the nodal point.

In all cases checked, the discrepancies between the manually computed values and the computer values was less than 10%.

5.2.4.1 Electrical Penetrations

Cartridge-type penetrations were used for all electrical conductors passing through the containment. This type of penetration is a hollow cylinder closed on both ends, through which the conductors pass. Each penetration cartridge provides a minimum of two pressure seals in series for each conductor. Each cartridge is provided with pressure connections to allow test pressurization for leak checking of the two pressure seals. There are a total of 110 electrical penetrations for the two units of the following types and quantities:

TYPE	QUANTITY
Power	
5 Kv	16
600 V	38
Control	24
Instrumentation	32

Figure 5.2-2 shows a typical electrical penetration.

The penetration sleeves which accommodate the electrical penetration cartridges are standard wall pipe of A333 Grade 6 carbon steel. Penetration sleeve ends were seal welded to weld rings, which are an integral part of the penetration cartridge.

Inspection and Testing

Electrical Penetrations - Prototype Tests

Prior to commencement of full production, a production prototype of the electrical penetrations listed in Table 5.2-3 successfully passed, in sequence, those tests indicated by X.

Upon completion of the above tests, each prototype successfully passed, a second time, the High Potential and Leakage test prescribed.

In addition to the prototype test listed in Table 5.2-3 all materials used in the penetrations were quality control inspected, tested and approved for service under operating and accident radiation dosages.

Electrical Penetrations Production Tests

Each completed electrical penetration successfully passed the following tests prior to shipment:

1. Leakage
2. Conductor Continuity Test
3. High Potential Test
4. Insulation Resistance.

5.2.4.2 Piping Penetrations

Piping penetrations are provided for all piping passing through the containment walls. The core pipe is contained within a sleeve which is welded to the containment liner. Several core pipes may pass through the same penetration assembly to minimize the number of penetrations required. In such cases, each core pipe is welded to the end plates in the penetration assembly. In the case of a pipe carrying a hot fluid, the core pipe may be insulated and cooling may be provided to limit the concrete temperature abutting the sleeve to 150°F.

The design ensures that, even under postulated accident conditions, potential resultant torsional, axial, bending and shear loads will not cause a breach of containment integrity. Penetrations were analyzed for the following conditions: a) Normal Operating Conditions; b) Transient Conditions; c) Seismic; d) Pipe Rupture (including consideration of the status of each pipe during the course of an accident). Loads on the penetration sleeve were combined following the principles in ASME Boiler and Pressure Vessel Code Section III. Penetrations were designed such that the rupture of connecting piping will not cause a loss of containment integrity.

Piping between the containment penetrations and the isolation valves outside the containment were designed in conformance with USAS B31.1 for design loads.

The main-steam pipe penetration assembly is similar to the hot pipe penetration illustrated in Figure 5.2-2. The core pipe within the penetration has a structural capability greater than that of the pipes welded to it. The penetration sleeve and core pipe are joined by a flued head which has a structural capability not less than the core pipe within the penetration assembly. The penetration sleeve in turn has adequate structural capability.

A complete thermal analysis was made of the penetration assembly to determine thermal insulation requirements to be used in conjunction with expanded plate-type coolers, to limit concrete temperature during normal operation to 150oF. Coolers were provided with redundant circuitry and capacity to maintain concrete temperature below 150oF with one circuit out of service. Thermal analysis to determine the time dependent limitations of penetration sleeve temperature limitations with regard to the containment liner and concrete was performed to cover conditions of loss of cooling water.

The thermal growth of the penetration sleeve and stress at the anchors and liner weld was considered in establishing temperature limitations.

The penetration assembly is anchored into the containment wall with a structural capability based upon Maximum Pipe Rupture Loads with regard to torsion, bending, shear, and jet thrust. Earthquake loads were considered.

The penetration assembly was designed to withstand any strains imposed by the liner.

The radial deformation imposed by the liner on the penetration sleeve was considered to be uniform around the circumference of the penetration sleeve and the moments and hoop stresses in the penetration sleeve then determined.

Stresses in the penetration were limited to the values stated in ASME Boiler and Pressure Vessel Code, Section III.

Sump Penetration

Two piping penetrations in the containment sump area are of the pipe and outer sleeve design. The outer sleeve is welded directly to the base of the liner. The weld to the liner is covered by a pressurization channel which is used to demonstrate liner integrity. The inner and outer pipes extend through the containment wall and are connected to an isolation valve and enclosure.

Fuel Transfer Penetration

A piping penetration, designated the fuel transfer tube penetration, is provided for fuel movement between the refueling canal in the containment and the transfer canal in the auxiliary building. The penetration consists of a stainless steel pipe installed inside a 24" pipe, as shown in detail on Figure 5.2-4. The inner pipe acts as the transfer tube and connects the containment refueling cavity with the fuel transfer canal in the auxiliary building.

The outer pipe is welded to the containment liner and provision was made for the employment of a seal ring for pressurizing welds essential to containment integrity. Bellows expansion joints were provided on the outer pipe to compensate for any differential movement between the inner and outer pipes and also between the containment and auxiliary building structures. These bellows do not serve as part of the containment pressure boundary.

Specification and Tests

Piping penetrations were designed to the intent of USAS B31.1 1967 Edition and N-Cases' (1955), and ASME Boiler and Pressure Vessel Code Section III 1968 Edition.

Material specifications for the piping penetrations are as follows;

Penetration Sleeve		ASTM A - 333 Gr 6
Penetration Reinforcing Rings		ASTM A - 442 Gr 60*
Penetration Sleeve Reinforcing		ASTM A - 442 Gr 60*
Bar Anchoring Rings, End Plates		
or Flued Heads		ASTM A - 442 GR 60*
		ASTM A - 350 LFI
		ASTM A -182 F 316 and F 304
Rolled Shapes		ASTM A - 442 Gr 60*
Core Pipe	Carbon Steel	ASTM A - 106 Grades B and C
		ASTM A - 155 KC 70 Class I
	Stainless Steel	ASTM A - 312 TP 304
		ASTM A - 358 Class I TP 304
		ASTM A - 376 TP 304 and TP 316
		ASTM A - 213 (Type 136)
		ASTM A - 249 (Type 316)

* or ASTM A 516 Gr. 70.

NDTT has been considered where required for the materials listed above. The piping penetration assemblies were tested, prior to installation, by pressurizing the annulus between the core pipe and sleeve for 30 minutes during which time the exterior was checked for leaks using a soap bubble solution. If any leakage was found, the assembly was repaired and the assembly retested. Following the soap bubble leakage test, the annulus was pressurized with a mixture of air and 20% by weight of freon gas. The assembly was then tested for leakage using a halogen leak detector with a sensitivity of 10-7 standard cc per second. A mass spectrometer examination was substituted for the halogen leak detection test where it was deemed required.

5.2.4.3 Equipment Hatches and Personnel Locks

Equipment Access Hatches

An equipment hatch with an inside diameter of 20'-0" has been provided to enable passage of large equipment and components into the containment during plant shutdown.

Design requirements include:

a. The materials for the equipment hatch conform to the requirements of ASTM A-300 Specifications. The minimum plate thickness is 1 inch.

   The design pressure is 12 psig, acting from the reactor side. The equipment hatch was fabricated and constructed as a Class "B" vessel in accordance with Section III of the ASME Code.

b. The hatch is equipped with double compression seals for leak tightness. A pressure connection has been provided between the seals for testing of the seals.

c. A removable floor has been provided capable of supporting a live load of 1,000 lbs/sq. ft. (If at any time the load being transferred throughout the equipment hatch exceeds the above load, the barrel of the hatch, both inside and outside the containment will be shored by means of temporary supports to prevent a structural failure of the body ring of the hatch).

Personnel Locks

Two personnel access locks have been provided, one of which penetrates the flat head of the equipment hatch. Each personnel lock is a welded steel assembly with a door at each end equipped with a double compressible seal to insure leak tightness of the lock. For details, see Figure 5.2-5.

Design requirements include:

a. The materials for the locks conforms to the requirements of ASTM A-516 Grade 70 firebox quality and ASTM A-300 Specifications. The minimum plate thickness is 3/8". The design pressure is 12 psig. The personnel locks were fabricated and constructed as Class "B" vessels in accordance with Section III of the ASME Code.

b. The doors, of the personnel locks, are interlocked so that one door cannot be opened unless the other is sealed.

c. Each door is equipped with a pressure valve for equalizing the pressure across each door. At no time can the equalizing valves on both doors be opened.

d. A test connection has been provided between the double compressible seals for allowing periodic leak testing of the seals.

e. All shafts penetrating the locks have double packing and a test connection has been provided for periodic pressure testing for leak tightness.

f. An emergency air supply has been provided to the inside of the lock. This connection was designed to permit periodic testing.

g. The locks have been equipped with pressure switches and with limit switches.

h. Indicating lights have been provided outside the lock at each door to indicate whether the opposite door is being operated.

The personnel locks were hydrostatically tested to 15 psig, i.e., 25% greater than the design pressure of 12 psig. Following the hydrostatic testing, the locks were tested for leaktightness by means of Freon-Air mixture, pressurized to the design pressure for 24 hours. All weld seams were checked with a Halogen leak detector.

<u>Accessibility Criteria</u>

Access to the containment during normal operation is limited and will be controlled in compliance with the limits set forth in 10CFR20.

For Class C piping, the following is provided for minimum isolation subsequent to an incident:

a) Incoming Lines: One check valve
b) Outgoing Lines: One auto-trip valve

Class D

Class D piping must remain in service after a hypothetical accident. Piping of the engineered safety features falls into this category.

For Class D piping the following is provided for minimum isolation subsequent to an incident.

a) Incoming Lines: One remote manual valve or a check valve.
b) Outgoing Lines: One remote manual valve

Class E

Class E piping is connected to a normally closed system outside of the containment, and is separated from the Reactor Coolant System and the containment atmosphere by a closed valve and/or a membrane barrier.

For Class E piping the following constitutes the minimum isolation provided.

All Lines: A normally closed manual valve inside or outside the containment.

5.4.2 CONTAINMENT ISOLATION SYSTEM DESIGN

The general design basis covering the number and location of isolation valves required to assure reactor containment integrity are given in Section 5.4.1. A summary of the major piping penetrations is given in Table 5.4-1. This table lists the number and types of isolation valves that are provided for the lines penetrating the containment. Valve positions during normal operation, shutdown, and incident conditions are also listed.

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 valved out and capped when not in use.

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

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
Non Essential Service Water to Instrument Room Ventilation Units	A	2½" (2)	In	Open	If Needed	Closed	-	2 Auto Trip	B	9.8-6	
Non Essential Service Water from Instrument Room Ventilation Units	A	2½" (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.	-	2 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
Post Accident Sampling System Supply (Gas)	D	1/2" (1)	Out	Closed	Closed	Int.	-	2 Auto Trip	A	9.6-2	13

11) May be put in service manually after incident
12) See Fig. 7.5-1 for a functional diagram of these instruments.
13) Connected to Containment Air Particulate and Radio Gas Detector Sample Line

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
Incore Flux Detection System	NA	8" (1)	-	Closed	If Needed	Closed	Blind Flange	Blind Flange	NA	-	13
Spare Penetrations	NA	18" (5) 6" (4)	-	Closed	Closed	Closed	Weld Cap	Weld Cap	NA	-	

13) Used for replacement of incore flux instrumentation thimbles.

N: Normal
S: shutdown
l: 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)

e. Maintain a maximum temperature of 100°F and a minimum temperature of 60°F in the Containment Instrumentation Room.

f. Purge the In-core Instrumentation Room atmosphere to the unit vent during periods of personnel access to this room.

g. Ensure that a reliable supply of cooling air is provided to the Control Rod Drive Mechanisms.

h. Reduce the concentration of airborne fission products (particulates, iodine and methyl iodine gases) which may be introduced into the containment atmosphere via leakage from the Reactor Coolant System (concurrent with 1 percent fuel cladding defects).

i. Aid in reduction of Containment pressure in the event of an accident. (See Chapter 14.)

j. Ensure that, in the case of a loss-of-coolant accident, any hydrogen that may be formed will not accumulate in pockets in excess of 4 percent (by volume).

k. Maintain concrete temperature below 150°F at the crane wall sleeves serving the RHR system when that system is operating.

In accordance with the Unit 2 Technical Specifications, the Unit 2 containment purge supply and exhaust and the instrument room purge supply and exhaust systems can only be operated during mode 5 or 6 operation without technical specification relief. The Unit 1 Technical Specifications allow use of these systems during any mode of operation.

5.5.3 SYSTEM DESCRIPTION

Containment Purge Supply and Exhaust System

One Containment Purge Supply and Exhaust System is supplied for each Containment structure so that, prior to entry, if required, radioactivity can be reduced to safe levels.

Purge air is supplied to the containment through two 16,000 CFM fans and their associated filters and heating coils. Purged air is exhausted through two 16,000 CFM capacity fans and absolute particulate filters to the unit vent where it is monitored before release to the atmosphere. The purge-air supply and exhaust fans and filters are located in the Auxiliary Building.

There are four air penetrations of the Containment associated with this system, a supply and an exhaust penetration into both the upper and lower compartment. Each penetration has two fail-closed isolation valves. (These valves are normally closed when the purge systems are not in operation.)

The Containment Purge Supply and Exhaust System has a total capacity of 32,000 CFM which affords approximately 1.5 air changes per hour.

The Containment Purge Supply and Exhaust System takes outside air through intake vents and passes it through medium-efficiency particulate filters (NBS Dust Spot Efficiency for atmospheric dust of 50%) and steam coils when necessary prior to discharge into the containment. The upper compartment purge exhaust plenum draws 11,000 CFM of air through inlets along the periphery of the refueling canal. The lower compartment purge exhaust plenum draws 21,000 CFM of air through inlets along the periphery of the reactor well cavity.

The Containment Purge Supply and Exhaust System serves to provide: 1) a means of reducing the radiation level in the containment to a safe value for containment entry, 2) a continuous airflow through the containment during refueling operations, and 3) heated air to the containment necessary for comfort of personnel working in the containment.

The Containment Purge Supply and Exhaust System is not normally operated. If, prior to containment entry, the containment radiation monitors indicate radiation levels in the containment area in excess of the appropriate Federal regulations for radiation exposure to an individual worker (per 10 CFR 20), and if it is determined that the radiation level within the containment is at a safe level for purging, then the Containment Purge Supply and Exhaust System is activated to reduce the radiation level within the containment to a safe value for containment entry.

In the unlikely event that radiation levels in the containment are too high for purging, the Containment Auxiliary Charcoal Filter System will be operated until radiation levels are low enough for purging. When the containment radiation level has been reduced to an acceptable point for purging, the Containment Purge Supply and Exhaust System isolation valves will be opened and the purge system will be actuated.

The Containment Purge Supply and Exhaust System fans are operated remotely from the Control Room, except that the Isolation valves close automatically upon a safety injection signal or a high containment radiation level.

During purge operations, the rate of purge can be controlled by the operator who has the option of operating any desired combination of the Containment Purge Supply and Exhaust System fans or by repositioning as necessary volume dampers (the volume dampers are located in the Auxiliary Building). Operation in this manner will prevent any undesirable containment pressure buildup, and will also provide a means of vacuum relief in the event of a negative containment pressure. Because containment pressures can be controlled entirely by operation of the Containment Purge Supply and Exhaust System during purging operations, there will be no need to use the Containment Pressure Relief System during Containment Purge.

Instrumentation Room Purge Supply and Exhaust System

The Containment Instrumentation Room is isolated from the general Containment atmosphere and has a separate and independent purge system consisting of a 1000 CFM supply unit and a 1000 CFM exhaust unit.

The supply unit draws outdoor air through an intake louver, passes it through a medium-efficiency particulate filter and electric blast coil heaters and discharges it into the Containment Instrumentation Room. The exhaust unit draws air from the Containment Instrumentation Room, passes it through both absolute particulate and charcoal filters and discharges it to the unit vent where it is monitored before release. This operation affords approximately 3-1/2 air changes per hour for the Containment Instrumentation Room.

Both the Containment Instrumentation Room purge supply and purge exhaust penetrations have two isolation valves similar in type and function to those provided for the Containment Purge Supply and Exhaust System.

Containment Pressure Relief System

Containment pressure relief is provided by a 1000 CFM exhaust unit composed of a fan, an absolute filter and a charcoal filter. This system is located in the Auxiliary Building. There is a single penetration of the containment barrier for this system with two isolation valves similar in type and function to those provided for the Containment Purge Supply and Exhaust System.

A flow diagram of the Containment Pressure Relief System is shown in Figure 5.5-2. The system fan draws containment atmosphere through a register in the upper compartment where, prior to discharge to the plant vent, it is passed through a filter unit containing both HEPA and charcoal filters. Additional features of the system design include two isolation valves, an automatically operated flow regulating damper which limits flow through the filters to 1000 cfm, a backdraft damper in the duct to the unit vent to prevent backflow from the unit vent into the containment, and a bypass path around the fan so that containment pressure relief can be provided in the event the pressure relief unit fan fails to start.

The system can be operated manually from the Control Room any time that containment pressure exceeds ambient. However, if the containment pressure should reach 0.2 psig, an alarm will sound in the control room to alert the operator to actuate the system.

The operator action required to actuate the system consists of opening the normally closed isolation valves and starting the fan motor. Such operator action will limit the containment internal pressure to less than 0.3 psig for normal atmospheric fluctuations.

If operation of the Containment Pressure Relief System is necessary, the containment atmosphere will always be exhausted through the charcoal and HEPA filters in the unit. This should be sufficient to prevent any adverse radioactivity from being exhausted to the environment. In any case, however, if during operation of the unit, a high radiation alarm sounds, the Containment Pressure Relief System isolation valves will automatically close. This will prevent any further release of adverse radioactivity to the environment.

The containment pressure relief system is intended for use only for normal operation when it is necessary to reduce internal containment pressure. It is not intended for use when the Containment Purge

Supply and Exhaust System is operating, since the Containment Purge Supply and Exhaust fans themselves provide the necessary means of controlling internal containment pressure.

<u>Upper Compartment Ventilation System</u>

The Upper Compartment Ventilation System consists of four free standing recirculating ventilating units (3 for normal operation, 1 standby). Each unit includes a 25,000 CFM fan, water cooling coils and electric blast coil heaters.

The water for the cooling coils is supplied by the Non-Essential Service Water System. Any three of the four units have sufficient cooling capacity to maintain the temperature below 100°F during design summer conditions. Water flow to the cooling coils is regulated by modulating air-operated valves located outside the containment. These valves are controlled by proportional thermostats located on the ventilation unit intakes. Maximum water flow is 80 gpm per unit.

Normally, three ventilation units operate continuously. Cooling is performed whenever the intake air temperature exceeds 90°F. The electric blast coil heaters are energized whenever the intake air temperature drops below 75°F.

<u>Lower Compartment Ventilation System</u>

The Lower Compartment Ventilation System is the largest of the Containment Ventilation Systems. It consists of four recirculation ventilation units composed of fans and water cooling coils, four booster fans for Control Rod Drive Mechanism ventilation, vent fans for reactor and pressurizer enclosure ventilation and associated duct work.

The four recirculation ventilation units are located in the annular space around the periphery of the lower chamber between the crane wall and the containment liner. Each unit is composed of water cooling coils and two 36,000 CFM fans. The intake to these units is connected via a duct penetration through the crane wall to air intakes from the top of the four steam generator enclosures, the Reactor Coolant Pump Motor areas and the discharges from the Control Rod Drive Mechanism vent fans. Air is drawn from the above stated heat sources, passed through the water cooling coils and discharged into the annular space. The cooled air re-enters the lower chamber via openings in the crane wall and through the pipe tunnel below the annular space which also has openings in the crane wall into the lower chamber.

The four recirculation units are split into pairs; two units in each of the two fan rooms. Normally, both fans of one unit and one of the fans of the second unit in a given room limit the average containment air temperature to 110°F. The water to the cooling coils is fed by the Non-Essential Service Water System. Water flow to each unit is modulated by an air-operated valve outside the containment which is controlled by a proportional thermostat in the recirculation unit intake. Maximum water flow per unit is 440 gpm.

There are four 20,000 CFM fans (1 standby) which draw air through the Control Rod Drive Mechanism shroud and discharge it into the intake ducts of the four lower compartment recirculation units. The four fans are located outside the primary shield of the reactor vessel and are all connected via a common intake header to the Control Rod Drive Mechanism ventilation shroud. There are redundant temperature sensors in the intake header which actuate an alarm in the Control Room in the event that the air temperature leaving the shroud exceeds the setpoint.

Two 3000 cfm booster fans draw air from the pipe tunnel and discharge it into the lower reactor cavity. This operation ensures a continuous flow of cool air at the base of the reactor vessel. Two 12,000 cfm fans (1 standby) draw air from the top of the pressurizer enclosure and discharge into the suction side of the lower containment ventilation system. This operation prevents heat buildup at the top of the enclosure. (The steam generator enclosures are ventilated by ducts which are also directly connected into the suction side of the lower containment ventilation system.)

Containment Instrumentation Room Ventilation System

The In-Core Instrumentation Room is an isolated sector of the lower compartment. The temperatures in the room are controlled by two free-standing, 9,600 cfm recirculation ventilation units (1 standby). Each unit is composed of a fan, water cooling coil and electric blast coil heaters. The water for coils is supplied by the Non-Essential Service Water System. Water flow is regulated in the same manner as for the upper compartment ventilation units. Maximum water flow per unit is 50 gpm. The Instrumentation Room is kept at a constant temperature of approximately 90°F during plant operation.

Containment Auxiliary Charcoal Filter System

This system consists of two 8000 cfm fan-filter units located in the lower containment compartment. Each unit contains both absolute particulate and charcoal filters, for reduction of fission product particulate activity which may be air-borne in the lower compartment.

The containment atmosphere is monitored for radioactivity during reactor power operation, and the number of auxiliary charcoal filter units in operation (none, 1 or 2) depends on the air-borne activity levels observed.

Containment Air Recirculation/Hydrogen Skimmer System

The Containment Air Recirculation/Hydrogen Skimmer System is the only safety related ventilation system within the containment. This system functions only in the event of a hi-hi containment pressure signal. It consists of two redundant independent systems which include fans, back draft dampers, valves, piping and ductwork.

Both Containment Air Recirculation Hydrogen Skimmer System Fans are located in the upper volume. The fans discharge, via the annular space between the crane wall and the Containment liner, into the lower compartment. The fans are provided with back draft dampers on the discharge to prevent backflow during initial blowdown.

Figure 5.5-2 shows the various components of this system and Figure 5.5-3 shows the recirculation flow patterns that are created by this system. The system includes provisions for providing both
1) general recirculation of containment atmosphere between the upper and lower compartments following a loss-of-coolant accident, and
2) preventing the improbable accumulation of hydrogen in restricted areas within the containment following a loss-of-coolant accident.

The potential areas of hydrogen pocketing are the top of the containment dome, and the lower compartment enclosures which include the three rooms in the annular space between the crane wall and the liner, the steam generator enclosures, and the pressurizer enclosure. Hydrogen pocketing is prevented by continuously drawing air out of the top of each of the above areas at such a rate as to limit the potential local hydrogen concentration to less than 4% by volume.

Each of the two independent systems fan has its own intake system composed of three separate headers. These headers draw 39,000 CFM from the upper compartment in the immediate vicinity of the fan, draw 1,000 CFM from the upper compartment at the top of the dome, and draw air from the potential hydrogen pockets in the lower compartment (this is the hydrogen skimmer header). Each header has volume control dampers in the line or at the air intake to balance flow. The hydrogen skimmer header is composed of two pipe branches, one which draws 500 CFM from the top of each double steam generator enclosure and pressurizer enclosure and one which draws 100 CFM from each of three rooms in the annular space. There is a normally closed, motor-operated hydrogen skimmer valve on each main hydrogen skimmer header to prevent ice condenser bypass during initial blowdown.

Ten minutes after receipt of a hi-hi containment pressure signal the Air Recirculation/Hydrogen Skimmer System fans start and the motor operated valves in the hydrogen skimmer header serving the lower compartment enclosures open. The total system design air flow per train is 41,800 SCFM.

Hot Sleeve Ventilation System

The hot sleeve ventilation system consists of two 3,000 CFM fans (1 standby, 1 active), which blow air through the three crane wall sleeve penetrations associated with the Residual Heat Removal System so that the temperature of the concrete at the sleeves will not exceed 150°F when the RHR system is operating.

5.5.4 DESIGN EVALUATION

The Containment Ventilation System provides adequate capacity to insure that proper temperatures are maintained in the various portions of the containment under operating and shutdown conditions in all types of weather.

The Containment Auxiliary Charcoal Filter System units will remove the airborne radioactivity that could result from leakage from the Reactor Coolant System (concurrent with 1 percent fuel cladding defects).

The Containment Purge Supply and Exhaust System provides the capability for changing the containment air prior to entry for refueling and maintenance. The Instrumentation Room can be purged independently of the balance of the containment so that entry may be achieved when necessary.

Containment Air Recirculation/Hydrogen Skimmer System

Each containment Air Recirculation/Hydrogen Skimmer System fan is designed to operate at a flow of 41,800 SCFM against a pressure drop through the fan inlet, across the back-draft damper and associated ductwork, and through the ice condenser from the lower volume to the upper volume. The hydrogen skimmer system is in parallel with a portion of the above flow circuit, and therefore is considered in the overall pressure drop against which the fan must operate. The static pressure drop against which the fan must operate is conservatively calculated as 4.175" w.g. and consists of the following:

Pressure drop through the ice condenser	2.075" w.g.
Pressure drop through damper across fan inlet (to assure adequate flow from the hydrogen skimmer system)	0.5" w.g.
Pressure drop through the backdraft damper and associated ductwork	1.6" w.g.
Total pressure drop	4.175" w.g.

The pressure drop through the ice condenser represents a conservative estimate of conditions in the ice condenser just after blowdown assuming that neither the intermediate nor top ice condenser doors are open and that just the vent area above the ice condenser is available for air recirculation. The actual pressure drop through the ice condenser following a loss-of-coolant accident will be much less than the above value, thus assuring that the flow capability of each Containment Air Recirculation/Hydrogen Skimmer System fan is greater than the required 41,800 scfm.

Containment Pressure Relief System

The Containment Pressure Relief System provides the capability for reducing the containment pressure by 0.1 psig in 8 minutes provided the atmospheric pressure remains constant. Based on extensive data taken as part of the site meteorology program, normally expected atmospheric fluctuations at the Cook Plant would not result in a change in atmospheric pressure of 0.1 psig in less than 40 minutes. Therefore, requiring the operator to actuate the Containment Pressure Relief System when the internal containment pressure reaches 0.2 psig assures that internal containment pressure will never reach 0.3 psig during normal plant operations.

The automatically operated air-operated damper in the Containment Pressure Relief System provides a means of maintaining a constant air flow through the charcoal and HEPA filters in the unit. Regulation of the flow in this manner will optimize the iodine absorption capability of the impregnated activated charcoal by limiting the face velocity through the charcoal filters, thus providing a minimum residence time of airflow of 0.25 seconds in each of the six 2-inch deep charcoal beds in this unit.

The HEPA/charcoal filters in the Containment Pressure Relief System have an exceedingly high capability for removal of both airborne particulate matter and airborne radioactive iodine. Both systems also have more than adequate capacity for retention of both particulates and iodine for the intended use of the system. The impregnated activated charcoal has a minimum absorption capability of 2.5 mg. of iodine for every gram of charcoal (total charcoal in this unit is a minimum of 37,100 grams). The single 24" x 24" x 12" HEPA filter is capable of holding at least 4 pounds of NBS Cottrell Precipitate Standardized Test Dust at a pressure drop of no more than 2.0 inches w.g.

5.5.5 INCIDENT CONTROL

In the event of an incident the two independent Containment Air Recirculation/Hydrogen Skimmer System fans automatically start after a 10 minute time delay after initiation of 2/4 hi-hi containment pressure signals. The operation of either fan ensures the reduction of the containment pressure to the limits described in Chapter 14.

At the same time the Air Recirculation/Hydrogen Skimmer fans start, the hydrogen skimmer valves in the two Containment Air Recirculation/ Hydrogen Skimmer headers open, thus causing the Air Recirculation/ Hydrogen Skimmer System fans to continuously purge all potential hydrogen pockets in the Containment.

All other Containment Ventilation Systems are not designed for operation during a loss of coolant accident.

The occurrence of a High Containment Radiation Signal from the upper compartment area or lower containment particulate/radiogas monitors will automatically trip the purge fans and close all ventilation system isolation control valves, thus isolating the Containment.

5.5.6 MALFUNCTION ANALYSIS

Sufficient redundancy exists in all recirculation ventilation systems to ensure a normal operation with one active component out of service.

The two filter cleanup units provide redundancy for small leakage rates. The Containment Purge Supply and Exhaust System is fitted with dual supply and exhaust fans. Simultaneous failure of a supply and an exhaust fan would result in an 80-minute purge rate.

The Containment Air Recirculation/Hydrogen Skimmer Systems are two 100% redundant systems, therefore the loss of either system or any component of either system will not impair system operation.

5.5.7 TESTS AND INSPECTION

All systems are inspected, tested and balanced upon installation. Charcoal and particulate filters are individually tested before shipment, upon installation and periodically thereafter as required. Replacement filters will be tested in the same manner.

The Containment Air Recirculation/Hydrogen Skimmer fans were tested during installation and are tested periodically to ensure proper functioning. The initial test of these fans were conducted at both no flow and full flow, verifying the fan capability to deliver the required amount of air. The periodic fan flow tests are conducted at no flow to assure that the fan is still operable.