Entergy

Entergy Nuclear Northeast
Entergy Nuclear Operations, Inc.
Indian Point 3 NPP
P.O. Box 308
Buchanan, NY 10511
Tel 914 736 8001 Fax 736 8012

Robert J. Barrett
Vice President, Operations–IP3

March 19, 2001
IPN-01-023

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

SUBJECT: **Indian Point 3 Nuclear Power Plant**
Docket No. 50-286
Improved Technical Specifications (ITS) Amendment Update to the Final Safety Analysis Report (FSAR)

REFERENCES:
1) 10 CFR 50.71(e)
2) NRC Letter dated February 27, 2001; "Indian Point Nuclear Generating Unit No. 3 – Issuance of Amendment for Conversion to Improved Standard Technical Specifications (TAC No. MA4359)," G. Wunder to M. Kansler.

Dear Sir:

As required by 10 CFR 50.71(e), (Reference 1) this letter transmits ten (10) copies of the amendment update to the Final Safety Analysis Report (FSAR) for the Indian Point 3 Nuclear Power Plant.

This update to the FSAR only incorporates changes of information being relocated from the current Indian Point 3 Technical Specifications into the FSAR resulting from the conversion to the Improved Technical Specifications on March 19, 2001 (Reference 2). There are no other plant or administrative changes being reflected in this update.

Any plant or administrative changes to the IP3 FSAR since December 31, 1999 will be incorporated into the next scheduled FSAR update, planned for six (6) months after our next refueling outage (RO11).

A053
1/10

No new commitments are being made by IP3 in this submittal. If you have questions, please contact Mr. Ken Peters.

Very truly yours,

Robert J. Barrett
Vice President - Operations
Indian Point 3 Nuclear Power Plant

State of New York
County of Westchester
Subscribed and sworn to before me

this 19 day of MARCH 2001

Notary Public

Christina Leitmann
Notary Public, State of New York
Registration #01LE5070946
Qualified In Putnam County
My Commission Expires Jan. 6, 2003

cc: U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Resident Inspector's Office
Indian Pint Unit 3
U.S. Nuclear Regulatory Commission
P.O. Box 337
Buchanan, NY 10511

Mr. G. Wunder, Project Manager
Project Directorate I
Division of Reactor Projects I/II
U.S. Nuclear Regulatory Commission
Mail Stop 8 C4
Washington, DC 20555

Mr. William M. Flynn, President
New York State Energy, Research, and Development Authority
286 Washington Avenue Extension
Albany, NY 12203-6399

STATUS OF FSAR PAGES, TABLES and FIGURES

This tabulation provides the latest status of all FSAR pages, tables and figures as of the revision number and date shown below. Maintenance of the FSAR in accordance with this status summary will assure an up-to-date version of the FSAR. In this tabulation, an F preceding the number in the "Page No." column indicates a figure, while a T denotes a table.

Page No.	Rev No.	Date
Chapter 1		
1.1-1	0	7/82
1.1-2	13	6/2000
1.1-3	5	6/2000
1.2-1	1	7/83
1.2-2	0	7/82
1.2-3	4	6/2000
1.2-4	1	7/90
1.2-5	3	6/99
1.2-6	4	6/2000
1.2-7	0	7/82
F1.2-1	0	7/82
F1.2-2	1	7/84
F1.2-3	3	7/94
F1.2-4	0	7/82
F1.2-5	5	7/94
F1.2-6	6	12/95
F1.2-7	3	12/97
F1.2-8	1	6/99
1.3-1	2	6/2000
1.3-2	1	6/2000
1.3-3	2	3/2001
1.3-4	0	7/82
1.3-5	0	7/82
1.3-6	0	7/82
1.3-7	0	7/82
1.3-8	2	12/97
1.3-9	2	6/2000
1.3-10	1	7/87
1.3-11	5	6/2000
1.3-12	3	3/2001
1.3-13	7	6/2000
1.3-14	5	6/2000
1.3-15	5	6/2000
1.3-16	2	7/91
1.3-17	2	6/2000
1.3-18	1	7/91
1.3-19	2	6/2000
1.3-20	1	6/99
1.3-21	2	6/2000
1.3-22	0	7/82
1.3-23	2	6/99
1.3-24	1	12/97
1.3-25	0	7/82
1.3-26	2	6/2000
1.3-27	1	7/88
1.3-28	0	7/82
1.3-29	0	7/82
1.3-30	0	7/82
1.3-31	0	7/82
1.3-32	1	6/99
1.3-33	0	7/82
1.3-34	0	7/82
1.3-35	0	7/82
1.3-36	0	7/82
1.3-37	1	6/99
1.3-38	0	7/82
1.3-39	1	12/95
1.3-40	5	6/2000
1.3-41	2	6/2000
1.3-42	0	7/82
1.3-43	0	7/82
1.3-44	0	7/82
1.3-45	0	7/82
1.3-46	0	7/82
1.3-47	0	7/82
1.3-48	0	7/82
1.3-49	2	6/2000
1.3-50	2	6/2000
1.3-51	6	3/2001
1.3-52	1	7/87
1.3-53	2	6/2000
1.3-54	2	7/88
1.3-55	1	7/87
1.3-56	3	6/99
1.3-57	1	7/87
1.3-58	1	7/87
1.3-59	1	7/87
1.3-60	1	7/87
1.3-61	2	6/2000
1.3-62	4	3/2001
1.3-63	1	7/87
1.3-64	3	6/2000
1.3-65	2	6/2000
1.3-66	0	7/82
1.3-67	2	3/2001
1.3-68	2	3/2001

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
1.3-69	1	7/88
1.3-70	0	7/82
1.3-71	0	7/82
1.3-72	2	6/2000
1.3-73	1	7/94
1.3-74	0	7/82
1.3-75	1	12/97
1.3-76	0	7/82
1.3-77	0	7/82
1.3-78	3	3/2001
1.3-79	1	12/95
1.3-80	1	12/95
1.3-81	1	3/2001
1.3-82	1	12/95
1.3-83	2	3/2001
1.3-84	1	6/99
1.3-85	1	12/97
1.3-86	3	6/99
1.3-87	3	6/99
1.3-88	2	6/2000
1.3-89	2	6/2000
1.3-90	2	6/2000
1.3-91	3	6/2000
1.3-92	1	6/2000
1.3-93	2	6/2000
1.3-94	3	3/2001
1.3-95	2	6/99
1.3-96	4	3/2001
1.3-97	2	3/2001
1.3-98	3	12/95
1.3-99	3	6/2000
1.3-100	4	6/2000
1.3-101	1	7/93
1.4-1	1	7/90
1.4-2	1	7/90
T1.4-1, Sh. 1	1	7/90
T1.4-1, Sh. 2	0	7/82
T1.4-1, Sh. 3	0	7/82
T1.4-1, Sh. 4	1	7/90
T1.4-1, Sh. 5	0	7/82
T1.4-1, Sh. 6	0	7/82
T1.4-1, Sh. 7	0	7/82
T1.4-1, Sh. 8	1	7/90
T1.4-1, Sh. 9	1	7/90
1.5-1	0	7/82
1.5-2	1	7/94
1.5-3	1	7/94
1.6-1	0	7/82
1.6-2	1	7/93
1.6-3	0	7/82
1.6-4	1	7/92
F1.6-1	0	7/82
F1.6-2	0	7/82
F1.6-3	0	7/82
	Chapter 2	
2.1-1	4	7/93
2.1-2	2	3/2001
2.1-3	0	7/82
2.1-4	0	7/82
2.2-1	3	7/94
2.2-2	1	7/94
F2.2-1	0	7/82
F2.2-2	1	7/84
F2.2-3	2	7/94
F2.2-4	0	7/82
2.3-1	0	7/82
2.4-1	1	7/92
2.4-2	2	7/92
2.4-3	2	7/92
2.4-4	Deleted	
T2.4-1	0	7/82
T2.4-2	1	7/92
T2.4-3	1	7/92
T2.4-4	1	7/92
T2.4-5	1	7/92
T2.4-6	1	7/92
T2.4-7	1	7/92
T2.4-8	1	7/92
T2.4-9	1	7/92
T2.4-10	1	7/92
T2.4-11	1	7/92
T2.4-12	1	7/92
T2.4-13	1	7/92
T2.4-14	1	7/92
T2.4-15	1	7/92
T2.4-16	1	7/92
T2.4-17	1	7/92
T2.4-18	1	7/92
T2.4-19	1	7/92
F2.4-1	0	7/82
F2.4-2	0	7/82
F2.4-3	0	7/82
F2.4-4	0	7/82
F2.4-5	0	7/82

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
2.7-2	0	7/82
2.7-3	0	7/82
2.7-4	0	7/82
F2.7-1	0	7/82
F2.7-2	0	7/82
F2.7-3	1	7/91
F2.7-4	0	7/82
2.8-1	1	7/91
2.8-2	1	7/91
2.8-3	2	7/93
2.8-4	1	7/93
T2.8-1	1	7/85
F2.8-1	0	7/82
2.9-1	1	7/85
2.9-2	5	3/2001
2.9-3	1	7/85
2.9-4	3	6/2000
2.9-5	Deleted	7/94
F2.9-1	Deleted	
	Chapter 3	
3.1-1	4	6/99
3.1-2	5	6/99
3.1-3	4	6/99
3.1-4	3	6/99
3.1-5	5	6/99
3.1-6	3	6/99
3.1-7	4	6/2000
3.1-8	7	6/99
3.1-9	7	3/2001
3.1-10	1	6/99
3.1-11	1	6/99
3.1-12	0	6/99
3.1-13	0	6/99
3.2-1	4	6/2000
3.2-2	3	6/2000
3.2-3	6	3/2001
3.2-4	3	6/2000
3.2-5	2	6/99
3.2-6	2	6/99
3.2-7	2	6/99
3.2-8	2	6/99
3.2-9	2	6/99
3.2-10	3	6/2000
3.2-11	3	6/99
3.2-12	3	6/2000
3.2-13	2	3/2001
3.2-14	2	6/99
3.2-15	1	6/99
3.2-16	1	6/99
3.2-17	1	6/99

Page No.	Rev No.	Date
3.2-18	2	6/99
3.2-19	2	6/99
3.2-20	1	6/99
3.2-21	1	6/99
3.2-22	1	6/99
3.2-23	1	6/99
3.2-24	1	6/99
3.2-25	1	6/99
3.2-26	1	6/99
3.2-27	1	6/99
3.2-28	1	6/99
3.2-29	2	6/99
3.2-30	2	6/99
3.2-31	2	6/99
3.2-32	3	6/99
3.2-33	3	6/99
3.2-34	2	6/99
3.2-35	2	6/99
3.2-36	2	6/99
3.2-37	2	6/99
3.2-38	3	6/99
3.2-39	2	6/99
3.2-40	3	6/99
3.2-41	3	6/99
3.2-42	4	6/2000
3.2-43	5	6/2000
3.2-44	3	6/99
3.2-45	3	6/99
3.2-46	3	6/99
3.2-47	3	6/99
3.2-48	3	6/99
3.2-49	3	6/99
3.2-50	3	6/99
3.2-51	4	6/2000
3.2-52	4	6/2000
3.2-53	3	6/99
3.2-54	3	6/99
3.2-55	3	6/99
3.2-56	3	6/99
3.2-57	3	6/99
3.2-58	3	6/99
3.2-59	6	6/99
3.2-60	2	6/99
3.2-61	2	6/99
3.2-62	1	6/99
3.2-63	1	6/99
3.2-64	3	6/99
3.2-65	0	6/99
3.2-66	0	6/99
3.2-67	0	6/99
3.2-68	0	6/99
3.2-69	0	6/99
3.2-70	0	6/99
3.2-71	0	6/99
3.2-72	1	6/2000

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
T3.2-1, Sh. 1	3	6/2000
T3.2-1, Sh. 2	3	6/2000
T3.2-1, Sh. 3	3	6/2000
T3.2-1, Sh. 4	1	6/2000
T3.2-2	2	6/2000
T3.2-3	2	6/2000
T3.2-4, Sh. 1	3	3/2001
T3.2-4, Sh. 2	2	6/2000
T3.2-4, Sh. 3	1	6/2000
T3.2-5, Sh. 1	2	6/2000
T3.2-5, Sh. 2	2	6/2000
T3.2-5, Sh. 3	1	6/2000
T3.2-6	0	7/92
F3.2-1	0	7/82
F3.2-2	0	7/82
F3.2-3	0	7/82
F3.2-4	0	7/82
F3.2-5	1	7/90
F3.2-5A	2	6/2000
F3.2-6	0	7/82
F3.2-7	0	7/82
F3.2-8	0	7/82
F3.2-9	0	7/82
F3.2-10	0	7/82
F3.2-11	0	7/82
F3.2-12	0	7/82
F3.2-13	0	7/82
F3.2-14	0	7/82
F3.2-15	0	7/82
F3.2-15A	0	7/90
F3.2-16	0	7/82
F3.2-17	0	7/82
F3.2-18	0	7/82
F3.2-19	0	7/82
F3.2-20	0	7/82
F3.2-21	0	7/82
F3.2-22	0	7/82
F3.2-23	0	7/82
F3.2-24	0	7/82
F3.2-25	0	7/82
F3.2-26	0	7/82
F3.2-27	0	7/82
F3.2-28	0	7/82
F3.2-29	0	7/82
F3.2-30	0	7/82
F3.2-31	0	7/82
F3.2-32	0	7/82
F3.2-33	0	7/82
F3.2-34	1	7/86
F3.2-34A	0	7/86
F3.2-35	0	7/82
F3.2-36	0	7/82
F3.2-37	Deleted	
F3.2-37A	1	12/97

Page No.	Rev No.	Date
F3.2-37B	0	7/90
F3.2-37C	0	12/97
F3.2-38	1	7/90
F3.2-39	2	6/2000
3.3-1	0	7/82
3.3-2	0	7/82
3.3-3	1	7/92
3.3-4	1	6/2000
3.3-5	3	6/2000
3.3-6	1	7/92
3.3-7	0	7/82
	Chapter 4	
4.1-1	3	7/90
4.1-2	0	7/82
4.1-3	0	7/82
4.1-4	0	7/82
4.1-5	0	7/82
4.1-6	1	7/88
4.1-7	0	7/82
4.1-8	0	7/82
4.1-9	0	7/82
4.1-10	2	3/2001
4.1-11	1	3/2001
4.1-12	1	3/2001
4.1-13	0	7/82
4.1-14	1	6/2000
4.1-15	0	7/82
4.1-16	0	7/82
T4.1-1	0	7/82
T4.1-2	0	7/82
T4.1-3	0	7/82
T4.1-4, Sh. 1	1	7/90
T4.1-4, Sh. 2	1	7/90
T4.1-5, Sh. 1	0	7/82
T4.1-5, Sh. 2	0	7/82
T4.1-6	0	7/82
T4.1-7	0	7/82
T4.1-8, Sh. 1	1	7/88
T4.1-8, Sh. 2	0	7/82
T4.1-9	1	7/90
4.2-1	3	12/97
4.2-2	2	7/94
4.2-3	3	6/2000
4.2-4	5	6/99
4.2-5	0	7/82
4.2-6	1	7/88
4.2-7	0	7/82
4.2-8	2	6/99
4.2-9	6	6/99

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
4.2-10	2	7/88
4.2-11	2	7/91
4.2-12	3	7/94
4.2-13	7	6/99
4.2-14	4	6/99
4.2-15	2	7/91
4.2-16	4	7/91
4.2-17	5	7/94
4.2-18	4	7/90
T4.2-1	1	7/90
T4.2-2	1	12/97
F4.2.1	Deleted	12/97
F4.2-2	2	7/91
F4.2-2A	4	7/94
F4.2-2B	9	6/2000
F4.2-3	1	12/95
F4.2-4	0	7/82
F4.2-5	1	7/90
F4.2-6	0	7/82
F4.2-7	1	6/2000
F4.2-8	0	7/82
F4.2-9	0	7/82
F4.2-10	1	6/99
F4.2-11	3	7/88
4.3-1	0	7/82
4.3-2	1	7/88
4.3-3	0	7/82
4.3-4	0	7/82
4.3-5	1	7/90
4.3-6	0	7/82
4.3-7	0	7/82
4.3-8	1	6/2000
4.3-9	0	7/82
4.3-10	2	6/2000
4.3-11	0	7/82
4.3-12	2	6/99
4.3-13	3	3/2001
4.3-14	3	3/2001
4.3-15	3	3/2001
4.3-16	1	7/90
4.3-17	2	7/90
4.3-18	1	7/90
4.3-19	1	7/90
4.3-20	2	6/2000
4.3-21	5	6/99
4.3-22	5	6/99
4.3-23	2	7/90
4.3-24	5	12/95
4.3-25	5	12/97
4.3-26	4	6/2000
4.3-27	1	6/2000
4.3-28	Deleted	

Page No.	Rev No.	Date
T4.3-1	0	7/82
T4.3-2	0	7/82
T4.3-3	1	7/90
T4.3-4	1	7/90
T4.3-5, Sh. 1	0	7/82
T4.3-5, Sh. 2	0	7/82
F4.3-1	0	7/82
F4.3-2	0	7/82
F4.3-3	0	7/82
F4.3-4	0	7/82
F4.3-5	0	7/82
F4.3-6	0	7/82
F4.3-7	0	7/82
F4.3-8	0	7/82
F4.3-9	0	7/82
4.4-1	7	6/99
4.4-2	6	6/99
4.4-3	3	3/2001
4.4-4	3	6/99
T4.4-1, Sh. 1	0	7/82
T4.4-1, Sh. 2	0	7/82
F4.4-1	1	7/84
4.5-1	1	7/87
4.5-2	2	6/2000
4.5-3	3	6/2000
4.5-4	5	6/2000
4.5-5	6	7/92
4.5-6	8	6/2000
4.5-7	2	6/2000
4.5-8	1	7/90
4.5-9	0	7/82
4.5-10	0	7/82
4.5-11	0	7/82
4.5-12	0	7/82
4.5-13	2	3/2001
4.5-14	2	3/2001
4.5-15	2	3/2001
4.5-16	4	7/90
4.5-17	0	7/82
4.5-18	0	7/82
T4.5-1, Sh. 1	1	7/90
T4.5-1, Sh. 2	1	7/90
T4.5-1, Sh. 3	1	7/90
F4.5-1	0	7/82
F4.5-2	0	7/82
4A-1	0	7/82
4A-2	1	6/2000
4A-3	0	7/82
4A-4	0	7/82
4A-5	0	7/82
4A-6	1	6/2000

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
4B-1	0	7/82
4B-2	0	7/82
4B-3	1	6/99
4B-4	1	6/99
4C-1	Deleted	
4C-2	Deleted	
4D-1	0	7/82
4D-2	0	7/82
4D-3	1	7/88
4D-4	2	7/90
4D-5	2	3/2001
4D-6	1	7/90
F4D-1	1	7/88
F4D-2	2	7/90
F4D-3	1	7/88
	Chapter 5	
5.1-1	0	7/82
5.1-2	0	7/82
5.1-3	1	12/95
5.1-4	1	6/2000
5.1-5	0	7/82
5.1-6	3	6/99
5.1-7	0	7/82
5.1-8	0	7/82
5.1-9	0	7/82
5.1-10	2	6/99
5.1-11	0	7/82
5.1-12	0	7/82
5.1-13	1	7/91
5.1-14	0	7/82
5.1-15	0	7/82
5.1-16	1	6/99
5.1-17	0	7/82
5.1-18	0	7/82
5.1-19	2	6/2000
5.1-20	1	6/99
5.1-21	0	7/82
5.1-22	0	7/82
5.1-23	0	7/82
5.1-24	1	6/99
5.1-25	1	6/99
5.1-26	2	6/2000
5.1-27	2	6/99
5.1-28	1	12/97
5.1-29	0	7/82

Page No.	Rev No.	Date
5.1-30	4	6/99
5.1-31	2	12/97
5.1-32	2	6/2000
5.1-33	0	7/82
5.1-34	2	6/99
5.1-35	0	7/82
5.1-36	1	12/95
5.1-37	0	7/82
5.1-38	1	6/99
5.1-39	4	6/2000
5.1-40	0	7/82
5.1-41	0	7/82
5.1-42	0	7/82
5.1-43	2	3/2001
5.1-44	2	3/2001
5.1-45	3	3/2001
5.1-46	0	7/82
T5.1-1	2	6/99
F5.1-1	0	7/82
F5.1-2	4	12/95
F5.1-3	4	6/99
F5.1-4	3	6/99
F5.1-5	2	7/92
F5.1-6	2	6/99
F5.1-7	3	12/95
F5.1-8	0	7/82
F5.1-9	0	7/82
F5.1-10	0	7/82
F5.1-11	0	7/82
F5.1-12	0	7/82
F5.1-13	0	7/82
F5.1-14	0	7/82
5.2-1	0	7/82
5.2-2	0	7/82
5.2-3	3	3/2001
5.2-4	4	6/99
5.2-5	6	3/2001
5.2-6	2	3/2001
5.2-7	3	3/2001
5.2-8	1	6/99
5.2-9	2	7/94
5.2-10	2	6/2000
5.2-11	0	6/2000
T5.2-1	4	7/94
T5.2-2	Deleted	6/99
T5.2-3, Sh. 1	7	6/2000
T5.2-3, Sh. 2	7	6/2000
T5.2-3, Sh. 3	8	6/2000
T5.2-3, Sh. 4	8	6/2000

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
T5.2-3, Sh. 5	6	6/2000
T5.2-3, Sh. 6	7	6/2000
T5.2-3, Sh. 7	6	6/2000
T5.2-3, Sh. 8	5	6/2000
T5.2-3, Sh. 9	0	6/2000
T5.2-3, Sh. 10	Deleted	6/99
T5.2-3, Sh. 11	Deleted	6/99
T5.2-3, Sh. 12	Deleted	6/99
T5.2-3, Sh. 13	Deleted	6/99
T5.2-3, Sh. 14	Deleted	6/99
T5.2-3, Sh. 15	Deleted	6/99
T5.2-3, Sh. 16	Deleted	6/99
T5.2-4, Sh. 1	3	6/99
T5.2-4, Sh. 2	6	6/2000
T5.2-4, Sh. 3	7	6/2000
F5.2-1	5	6/99
F5.2-2	3	6/99
F5.2-3	3	6/99
F5.2-4	4	6/99
F5.2-5	2	6/99
F5.2-6	2	6/99
F5.2-7	7	6/99
F5.2-8	4	6/99
F5.2-9	3	6/99
F5.2-10	3	6/99
F5.2-11	3	6/99
F5.2-12	3	6/99
F5.2-13	3	6/99
F5.2-14	2	6/99
F5.2-15	3	6/99
F5.2-15A	0	6/99
F5.2-15B	0	6/99
F5.2-16	3	6/99
F5.2-16A	0	6/99
F5.2-17	5	6/2000
F5.2-18	2	7/91
F5.2-19	2	6/99
F5.2-20	3	6/99
F5.2-21	1	6/99
F5.2-22	3	6/99
F5.2-23	Deleted	7/93
F5.2-24	Deleted	7/93
F5.2-25	2	6/99
F5.2-26	2	6/2000
F5.2-27	2	6/99
F5.2-27A	1	6/99
F5.2-28	4	6/99
F5.2-29	4	6/99
5.3-1	2	6/2000
5.3-2	5	6/2000
5.3-3	4	6/2000
5.3-4	4	12/95
5.3-5	0	7/82

Page No.	Rev No.	Date
T5.3-1, Sh. 1	4	6/99
T5.3-1, Sh. 2	1	6/2000
T5.3-1, Sh. 3	2	6/2000
5.4-1	0	7/82
5.4-2	0	7/82
5.4-3	1	7/83
5.4-4	0	7/82
5.4-5	0	7/82
T5.4-1	0	7/82
T5.4-2	1	7/83
T5.4-4	0	7/82
F5.4-1	4	6/2000
F5.4-2	1	6/99
5.5-1	6	6/2000

Report – Sept. 1970: Containment Design, pages 5A-1 to 5A-101

Chapter 6

Page No.	Rev No.	Date
6.1-1	2	6/99
6.1-2	2	6/99
6.1-3	3	6/2000
6.1-4	1	6/2000
6.1-5	0	7/82
6.1-6	1	7/84
6.1-7	0	7/82
6.1-8	3	7/90
6.1-9	2	7/89
6.2-1	1	7/93
6.2-2	1	7/93
6.2-3	1	7/93
6.2-4	2	6/99
6.2-5	4	6/2000
6.2-6	5	12/97
6.2-7	7	6/2000
6.2-8	4	6/2000
6.2-9	5	3/2001
6.2-10	7	3/2001
6.2-11	8	3/2001
6.2-12	6	3/2001
6.2-13	7	6/2000
6.2-14	6	6/2000
6.2-15	7	6/2000
6.2-16	4	6/2000
6.2-17	2	12/95
6.2-18	7	6/2000
6.2-19	5	3/2001
6.2-20	5	3/2001

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
6.2-21	4	6/2000
6.2-22	3	7/93
6.2-23	2	7/93
6.2-24	4	6/2000
6.2-25	4	6/2000
6.2-26	3	6/2000
6.2-27	4	6/2000
6.2-28	3	6/2000
6.2-29	1	7/93
6.2-30	1	7/93
6.2-31	1	7/93
6.2-32	2	12/97
6.2-33	3	6/2000
6.2-34	3	6/2000
6.2-35	6	6/2000
6.2-36	4	3/2001
6.2-37	3	12/97
6.2-38	5	6/99
6.2-39	2	12/97
6.2-40	3	6/2000
6.2-41	4	3/2001
6.2-42	4	3/2001
6.2-43	6	3/2001
6.2-44	2	6/2000
6.2-45	1	7/93
6.2-46	3	6/2000
6.2-47	Deleted	7/93
6.2-48	Deleted	7/93
T6.2-1	2	7/93
T6.2-2	4	7/93
T6.2-3	2	6/2000
T6.2-4	3	12/95
T6.2-5, Sh. 1	3	6/2000
T6.2-5, Sh. 2	5	6/2000
T6.2-6	2	7/92
T6.2-7, Sh. 1	3	6/2000
T6.2-7, Sh. 2	2	12/97
T6.2-7, Sh. 3	1	7/93
T6.2-7, Sh. 4	1	7/93
T6.2-7, Sh. 5	0	7/93
T6.2-7, Sh. 6	0	7/93
T6.2-7, Sh. 7	4	6/2000
T6.2-8, Sh. 1	3	6/2000
T6.2-8, Sh. 2	1	7/93
T6.2-9, Sh. 1	2	12/95
T6.2-9, Sh. 2	3	6/2000
T6.2-10	2	6/99
T6.2-11, Sh. 1	2	12/95
T6.2-11, Sh. 2	1	7/93
T6.2-12, Sh. 1	0	7/82
T6.2-12, Sh. 2	0	7/82
T6.2-12, Sh. 3	0	7/82
T6.2-12, Sh. 4	0	7/82
T6.2-13, Sh. 1	7	6/2000
T6.2-13, Sh. 2	5	6/99
T6.2-13, Att.	1	12/97

Page No.	Rev No.	Date
F6.2-1	Deleted	
F6.2-1A	7	6/2000
F6.2-1B	9	6/2000
F6.2-2	2	12/95
F6.2-3	1	7/93
F6.2-4	2	12/97
6.3-1	0	7/82
6.3-2	0	7/82
6.3-3	2	6/99
6.3-4	1	7/90
6.3-5	2	6/99
6.3-6	4	6/2000
6.3-7	2	6/2000
6.3-8	3	6/2000
6.3-9	1	7/90
6.3-10	2	6/2000
6.3-11	1	12/95
6.3-12	1	6/2000
6.3-13	4	6/99
6.3-14	2	12/95
6.3-15	5	3/2001
6.3-16	2	3/2001
6.3-17	0	7/82
T6.3-1	0	7/82
T6.3-2	2	6/2000
T6.3-3	1	7/93
T6.3-4, Sh. 1	1	6/2000
T6.3-4, Sh. 2	1	6/2000
T6.3-4, Sh. 3	0	7/82
T6.3-5	0	7/82
F6.3-1	0	7/82
6.4-1	0	7/82
6.4-2	1	7/94
6.4-3	0	7/82
6.4-4	5	6/99
6.4-5	2	6/99
6.4-6	3	6/2000
6.4-7	3	6/2000
6.4-8	2	6/2000
6.4-9	3	6/2000
6.4-10	4	6/2000
6.4-11	1	6/2000
6.4-12	2	6/2000
6.4-13	3	6/2000
6.4-14	5	6/2000
6.4-15	3	6/2000
6.4-16	6	6/2000
6.4-17	2	3/2001
6.4-18	4	3/2001
6.4-19	2	3/2001

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date	Page No.	Rev No.	Date
6.4-20	3	6/99	6.7-12	4	6/2000
6.4-21	5	3/2001	6.7-13	3	12/95
6.4-22	2	7/91	6.7-14	1	7/93
T6.4-1	1	7/94	6.7-15	3	6/2000
T6.4-2	2	6/2000	6.7-16	3	6/2000
F6.4-1	1	6/99	6.7-17	3	6/2000
F6.4-2	9	6/2000	6.7-18	3	6/2000
F6.4-3	3	6/99	6.7-19	3	6/2000
F6.4-4	0	7/82	6.7-20	2	6/99
			6.7-20A	0	6/99
6.5-1	1	6/99	6.7-21	3	6/2000
6.5-2	2	6/2000	6.7-22	3	6/2000
6.5-3	1	12/95	6.7-23	0	6/99
6.5-4	3	6/2000	6.7-24	Deleted	7/93
6.5-5	0	7/82	6.7-25	Deleted	7/93
6.5-6	2	6/99	6.7-26	Deleted	7/93
6.5-7	3	6/99	T6.7-1	0	7/82
6.5-8	0	7/82	T6.7-2	1	12/97
T6.5-1	0	7/82	T6.7-3	0	6/99
T6.5-2	0	7/82			
T6.5-3	0	7/82	6.8-1	2	12/95
F6.5-1	9	6/2000	6.8-2	2	3/2001
F6.5-2	0	7/82	6.8-3	4	3/2001
			6.8-4	2	3/2001
6.6-1	1	12/95	6.8-5	3	3/2001
6.6-2	5	6/2000	6.8-6	3	6/2000
6.6-3	2	12/97	6.8-7	Deleted	7/93
6.6-4	1	12/95	6.8-8	Deleted	7/93
6.6-5	3	6/99	6.8-9	Deleted	7/93
6.6-6	3	3/2001	6.8-10	Deleted	7/93
6.6-7	3	12/97	6.8-11	Deleted	7/93
T6.6-1	1	7/90	6.8-12	Deleted	7/93
T6.6-2	0	7/82	T6.8-1	0	7/93
T6.6-3	1	12/97	F6.8-1	1	7/93
T6.6-4	0	7/82	F6.8-2	7	6/2000
F6.6-1	9	12/97	F6.8-2A	2	6/2000
6.7-1	1	7/93	6A-1	1	6/99
6.7-2	1	7/93	6A-2	1	6/99
6.7-3	2	7/93	6A-3	1	6/99
6.7-4	4	6/2000	6A-4	1	6/99
6.7-5	2	6/2000	6A-5	Deleted	6/99
6.7-6	2	6/2000	6A-6	Deleted	6/99
6.7-7	1	7/93	6A-7	Deleted	6/99
6.7-8	4	6/2000	6A-8	Deleted	6/99
6.7-9	3	6/2000	6A-9	Deleted	6/99
6.7-10	1	7/93	6A-10	2	6/99
6.7-11	3	6/2000	6A-11	Deleted	7/93

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
7.3-13	6	6/2000
7.3-14	3	6/2000
7.3-15	2	6/99
7.3-16	1	12/97
F7.3-1	1	7/90
7.4-1	2	6/2000
7.4-2	2	6/2000
7.4-3	4	6/2000
7.4-4	3	7/90
7.4-5	4	6/99
7.4-6	4	6/2000
7.4-7	2	6/2000
7.4-8	2	6/99
7.4-9	1	7/88
T7.4-1	0	7/82
F7.4-1	2	6/2000
F7.4-2	0	7/82
F7.4-3	3	6/2000
F7.4-4	2	6/2000
7.5-1	2	12/95
7.5-2	3	12/95
7.5-3	3	12/95
7.5-4	4	6/2000
7.5-5	2	3/2001
T7.5-1, Sh. 1	0	6/99
T7.5-1, Sh. 2	1	3/2001
T7.5-1, Sh. 3	0	6/99
T7.5-1, Sh. 4	0	6/99
T7.5-1, Sh. 5	0	6/99
T7.5-1, Sh. 6	0	6/99
T7.5-1, Sh. 7	0	6/99
T7.5-1, Sh. 8	0	6/99
T7.5-1, Sh. 9	0	6/99
T7.5-1, Sh. 10	0	6/99
T7.5-1, Sh. 11	2	3/2001
T7.5-1, Sh. 12	0	6/99
T7.5-1, Sh. 13	0	6/99
T7.5-1, Sh. 14	0	6/99
T7.5-1, Sh. 15	0	6/99
T7.5-1, Sh. 16	0	6/99
T7.5-1, Sh. 17	0	6/99
T7.5-1, Sh. 18	0	6/99
T7.5-1, Sh. 19	0	6/99
F7.5-1	2	6/2000
F7.5-2	0	7/88
7.6-1	3	7/93
7.6-2	1	7/90
7.6-3	1	3/2001
7.6-4	1	3/2001

Page No.	Rev No.	Date
F7.6-1	3	6/2000
F7.6-2	0	7/82
F7.6-3	1	6/2000
F7.6-4	1	6/2000
7.7-1	4	3/2001
7.7-2	3	3/2001
7.7-3	4	6/2000
7.7-4	2	6/2000
7.7-5	1	6/2000
7.7-6	3	7/91
7.7-7	3	6/2000
7.7-8	3	6/2000
7.7-9	3	12/97
7.7-10	3	6/99
7.7-11	2	6/99
F7.7-1	6	6/2000
F7.7-2	6	6/2000
7.8-1	0	7/82
7.9-1	1	3/2001
	Chapter 8	
8.1-1	2	7/94
8.1-2	1	7/94
8.1-3	4	6/2000
8.2-1	2	6/99
8.2-2	4	6/99
8.2-3	2	12/95
8.2-4	2	12/95
8.2-5	2	12/97
8.2-6	6	6/99
8.2-7	1	7/86
8.2-8	1	7/86
8.2-9	3	6/2000
8.2-10	3	12/95
8.2-11	4	12/95
8.2-12	6	6/2000
8.2-13	6	6/2000
8.2-14	5	6/2000
8.2-15	6	6/2000
8.2-16	7	3/2001
8.2-17	Deleted	12/97
8.2-18	Deleted	12/97
8.2-19	Deleted	12/97
8.2-20	3	6/2000
8.2-21	4	3/2001
8.2-22	0	7/91
8.2-23	0	6/2000

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
T8.2-1	Deleted	7/91
T8.2-1A	1	12/97
T8.2-1B	Deleted	6/2000
T8.2-1C	Deleted	6/2000
T8.2-1D	Deleted	6/2000
T8.2-1E	Deleted	6/2000
T8.2-1F	Deleted	6/2000
T8.2-2	5	6/2000
T8.2-3, Sh. 1	3	7/91
T8.2-3, Sh. 2	3	7/91
T8.2-3, Sh. 3	4	7/91
T8.2-3, Sh. 4	4	7/91
F8.2-1	3	6/2000
F8.2-2	7	6/2000
F8.2-3	4	6/2000
F8.2-4	11	6/2000
F8.2-5	Deleted	12/95
F8.2-5A	2	6/2000
F8.2-5B	1	6/2000
F8.2-6	12	6/2000
F8.2-7	9	6/2000
F8.2-8	9	6/99
F8.2-9	13	12/97
8.3-1	1	3/2001
8.4-1	2	6/2000
8.4-2	2	6/2000
8.4-3	4	12/95
8.4-4	0	6/2000
F8.4-1	3	6/2000
8.5-1	1	3/2001
	Chapter 9	
9.1-1	2	6/2000
9.1-2	0	7/82
9.2-1	1	7/88
9.2-2	2	6/2000
9.2-3	2	12/95
9.2-4	1	6/2000
9.2-5	2	6/99
9.2-6	1	7/88
9.2-7	1	7/88
9.2-8	1	7/88
9.2-9	1	7/88
9.2-10	2	12/97
9.2-11	3	6/2000
9.2-12	2	6/2000
9.2-13	1	7/88
9.2-14	1	7/88
9.2-15	4	6/2000

Page No.	Rev No.	Date
9.2-16	2	6/2000
9.2-17	1	7/88
9.2-18	1	7/88
9.2-19	2	7/91
9.2-20	2	7/93
9.2-21	5	6/2000
9.2-22	4	6/2000
9.2-23	3	6/99
9.2-24	2	12/95
9.2-25	4	12/97
9.2-26	5	6/2000
9.2-27	4	6/2000
9.2-28	3	12/95
9.2-29	3	12/95
9.2-30	3	12/95
9.2-31	3	12/95
9.2-32	4	6/2000
9.2-33	1	6/99
9.2-34	Deleted	7/88
9.2-35	Deleted	7/88
9.2-36	Deleted	7/88
9.2-37	Deleted	7/88
9.2-38	Deleted	7/88
T9.2-1	1	7/88
T9.2-2	2	6/2000
T9.2-3, Sh. 1	1	7/88
T9.2-3, Sh. 2	2	7/94
T9.2-4	0	7/82
T9.2-5	0	7/82
T9.2-6, Sh. 1	1	6/2000
T9.2-6, Sh. 2	0	7/82
T9.2-6, Sh. 3	1	12/97
T9.2-7	1	6/2000
F9.2-1	11	6/2000
F9.2-2	9	6/2000
F9.2-3	Deleted	7/88
F9.2-3	Deleted	7/88
9.3-1	1	6/2000
9.3-2	2	12/97
9.3-3	3	6/2000
9.3-4	3	12/97
9.3-5	5	3/2001
9.3-6	4	3/2001
9.3-7	4	3/2001
9.3-8	3	3/2001
9.3-9	0	7/82
9.3-10	1	7/91
9.3-11	1	7/91
9.3-12	6	6/99
9.3-13	3	6/2000
9.3-14	2	6/99
9.3-15	4	3/2001
9.3-16	3	3/2001

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
9.3-17	0	6/2000
T9.3-1, Sh. 1	2	7/93
T9.3-1, Sh. 2	4	12/95
T9.3-1, Sh. 3	3	6/2000
T9.3-2, Sh. 1	3	7/93
T9.3-2, Sh. 2	2	7/93
T9.3-2, Sh. 3	Deleted	7/91
T9.3-3, Sh. 1	3	6/99
T9.3-3, Sh. 2	4	6/2000
T9.3-3, Sh. 3	0	7/82
T9.3-3, Sh. 4	0	7/82
T9.3-3, Sh. 5	1	12/97
T9.3-3, Sh. 6	0	7/82
T9.3-4	0	7/82
T9.3-5	1	7/91
F9.3-1	7	6/2000
F9.3-2	Deleted	
F9.3-2A	9	6/2000
F9.3-2B	8	6/2000
9.4-1	1	6/2000
9.4-2	2	6/2000
9.4-3	4	6/2000
9.4-4	3	6/2000
9.4-5	3	6/2000
9.4-6	2	3/2001
9.4-7	7	12/97
9.4-8	7	3/2001
9.4-9	1	3/2001
T9.4-1	0	7/82
T9.4-2	3	7/94
T9.4-2, Sh. 2	Deleted	794
T9.4-3	0	7/82
T9.4-4	0	12/97
F9.4-1	10	6/2000
F9.4-2	1	7/85
9.5-1	4	6/2000
9.5-2	1	12/97
9.5-3	1	12/97
9.5-4	3	3/2001
9.5-5	2	3/2001
9.5-6	1	6/2000
9.5-7	3	6/2000
9.5-8	4	6/99
9.5-9	3	3/2001
9.5-10	5	3/2001
9.5-11	1	7/91
9.5-12	2	6/2000
9.5-13	2	6/2000
9.5-13A	1	6/2000
9.5-14	3	3/2001
9.5-15	3	3/2001
9.5-16	5	3/2001

Page No.	Rev No.	Date
9.5-17	1	12/97
9.5-18	2	6/2000
9.5-19	2	6/99
9.5-20	1	7/91
9.5-21	2	3/2001
9.5-22	2	3/2001
9.5-23	2	6/2000
9.5-24	5	3/2001
9.5-25	0	3/2001
T9.5-1	5	6/99
T9.5-1, Sh. 2	Deleted	7/94
F9.5-1	1	12/97
F9.5-2	3	12/97
F9.5-2A	0	6/99
F9.5-2B	0	6/99
F9.5-2C	0	6/99
F9.5-3	0	7/82
F9.5-4	1	3/2001
F9.5-5	2	3/2001
F9.5-6	0	7/82
F9.5-7	0	7/82
F9.5-8	0	7/82
F9.5-9	1	12/97
9.6-1	7	12/97
9.6-2	7	6/2000
9.6-3	7	6/2000
9.6-4	6	6/2000
9.6-5	3	7/90
9.6-6	4	7/91
9.6-7	6	6/99
9.6-8	8	6/2000
9.6-9	3	7/94
9.6-10	7	6/2000
9.6-11	3	12/97
9.6-12	7	6/99
9.6-13	4	12/97
9.6-14	5	6/2000
9.6-15	6	6/2000
9.6-16	5	6/99
9.6-17	7	6/99
9.6-18	7	6/2000
9.6-19	7	6/99
9.6-20	6	6/99
9.6-21	7	6/2000
9.6-22	5	6/2000
9.6-23	4	6/99
9.6-24	4	6/2000
9.6-25	4	6/99
9.6-26	3	12/97
9.6-27	5	12/97
9.6-28	3	12/97
9.6-29	6	6/2000
9.6-30	5	6/99
9.6-31	4	6/99
9.6-32	7	6/2000

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
9.6-33	3	12/97
9.6-34	3	12/97
9.6-35	4	6/99
9.6-36	3	6/2000
9.6-37	6	6/2000
9.6-38	7	6/2000
9.6-39	8	6/2000
9.6-40	6	6/2000
9.6-41	6	12/95
9.6-42	6	6/2000
9.6-43	5	6/2000
9.6-44	6	6/2000
9.6-45	3	12/95
9.6-46	1	7/89
9.6-47	4	6/2000
9.6-48	4	6/2000
9.6-49	3	7/93
9.6-50	2	12/97
T9.6-1, Sh. 1	7	6/2000
T9.6-1, Sh. 2	7	6/2000
T9.6-2	5	6/2000
T9.6-2A, Sh. 1	4	6/2000
T9.6-2A, Sh. 2	3	6/2000
T9.6-2A, Sh. 3	4	6/2000
T9.6-2A, Sh. 4	3	6/2000
T9.6-3	6	6/2000
T9.6-4	Deleted	7/88
T9.6-5	Deleted	7/88
T9.6-6	Deleted	7/88
F9.6-1	Deleted	7/88
F9.6-1A	8	6/2000
F9.6-1B	8	6/2000
F9.6-1C	6	6/2000
F9.6-2	Deleted	7/88
F9.6-2A	4	12/97
F9.6-2B	4	12/97
F9.6-3	4	12/97
F9.6-4	Deleted	7/88
F9.6-4A	4	12/97
F9.6-4B	4	12/97
F9.6-5	Deleted	7/88
F9.6-6	Deleted	7/88
F9.6-7	Deleted	7/88
F9.6-8	Deleted	7/88
F9.6-9	Deleted	7/88
F9.6-9A	10	6/2000
F9.6-9B	8	6/2000
F9.6-10	Deleted	7/88
F9.6-11	Deleted	7/88
F9.6-12	4	6/2000
F9.6-13	Deleted	12/97
F9.6-13A	1	6/2000
F9.6-13B	2	6/2000

Page No.	Rev No.	Date
F9.6-14	Deleted	12/95
F9.6-15	5	6/99
F9.6-16	8	6/2000
F9.6-17	10	6/2000
9.7-1	0	7/82
9.7-2	1	6/2000
9.8-1	4	6/99
9.8-2	3	6/99
9.8-3	1	12/95
T9.8-1	0	7/82
9.9-1	4	6/2000
9.9-2	4	6/2000
9.9-3	2	12/97
9.9-4	3	6/2000
9.9-5	5	6/99
9.9-6	5	6/2000
9.9-7	4	6/99
F9.9-1	4	6/2000
F9.9-2	Deleted	6/2000
9.10-1	2	6/99
9.11-1	1	7/87

Chapter 10

Page No.	Rev No.	Date
10.1-1	2	6/2000
10.1-2	2	7/91
T10.1-1, Sh. 1	0	7/82
T10.1-1, Sh. 2	0	7/82
10.2-1	3	6/2000
10.2-2	2	3/2001
10.2-3	1	7/88
10.2-4	2	7/91
10.2-5	3	7/93
10.2-6	3	12/97
10.2-7	2	7/90
10.2-8	1	7/88
10.2-9	3	7/91
10.2-10	3	7/93
10.2-11	7	6/2000
10.2-12	6	12/95
10.2-13	6	12/95
10.2-14	4	7/94
10.2-15	5	6/99
10.2-16	5	6/99
10.2-17	4	794
10.2-18	3	7/91
10.2-19	4	6/2000

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date	Page No.	Rev No.	Date
10.2-20	5	12/95	F10.2-21	0	7/82
10.2-21	3	7/91	F10.2-22	0	7/82
10.2-22	4	6/99	F10.2-23	0	7/82
10.2-23	4	6/2000	F10.2-24	9	6/2000
10.2-24	5	7/91	F10.2-25	2	6/2000
10.2-25	6	6/99	F10.2-26	5	6/2000
10.2-26	8	6/99	F10.2-27	5	6/99
10.2-27	8	6/2000	F10.2-28	5	12/97
10.2-28	7	6/2000	F10.2-29	0	7/82
10.2-29	6	6/2000	F10.2-30	1	7/90
10.2-30	6	6/2000	F10.2-31	Deleted	7/90
10.2-31	7	3/2001	F10.2-32	1	7/90
10.2-32	6	3/2001	F10.2-33	0	7/82
10.2-33	6	6/99	F10.2-34	0	7/82
10.2-34	1	12/95	F10.2-35	0	7/82
T10.2-1	2	6/99	F10.2-36	0	7/82
T10.2-2	0	7/82	F10.2-37	0	7/82
T10.2-3	0	7/82	F10.2-38	1	6/99
T10.2-4	0	7/82	F10.2-39	1	6/99
T10.2-5	0	7/82	F10.2-40	2	7/94
T10.2-6	0	7/82	F10.2-41	1	6/99
T10.2-7	0	7/82	F10.2-42	2	6/99
T10.2-8	2	12/97	F10.2-43	1	6/99
F10.2-1	0	7/82	F10.2-44	1	7/90
F10.2-2	10	6/2000	F10.2-45	1	6/99
F10.2-3	Deleted	7/87	F10.2-46	1	12/97
F10.2-3A	9	6/2000	F10.2-47A	8	6/2000
F10.2-3B	7	12/97	F10.2-47B	7	6/2000
F10.2-3C	Deleted	7/88	F10.2-48	6	6/2000
F10.2-4	11	6/2000			
F10.2-5	Deleted	7/88	10.3-1	0	7/82
F10.2-5A	7	6/2000	10.3-2	1	12/97
F10.2-5B	4	6/99	10.3-3	0	7/82
F10.2-6	Deleted	7/88	T10.3-1	1	6/2000
F10.2-7	Deleted	7/88			
F10.2-7A	7	6/99	10.4-1	0	7/82
F10.2-7B	4	12/97			
F10.2-8	Deleted	7/88		Chapter 11	
F10.2-8A	6	6/2000			
F10.2-8B	7	6/99	11.1-1	2	7/89
F10.2-9	9	6/2000	11.1-2	5	6/2000
F10.2-10	9	12/97	11.1-3	6	6/2000
F10.2-11	7	12/97	11.1-4	5	3/2001
F10.2-12	8	6/2000	11.1-5	6	3/2001
F10.2-13	9	6/99	11.1-6	Deleted	7/89
F10.2-14	0	7/82	11.1-7	3	6/99
F10.2-15	0	7/82	11.1-8	2	12/97
F10.2-16	0	7/82	11.1-9	6	6/99
F10.2-17	0	7/82	11.1-10	5	12/97
F10.2-18	0	7/82	11.1-11	5	6/99
F10.2-19	0	7/82	11.1-12	5	6/99
F10.2-20	0	7/82	11.1-13	6	3/2001

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
11.1-14	4	6/99
11.1-15	6	3/2001
11.1-16	5	3/2001
T11.1-1, Sh. 1	1	7/93
T11.1-1, Sh. 2	1	7/93
T11.1-1, Sh. 3	1	7/93
T11.1-1, Sh. 4	1	7/93
T11.1-2	2	7/89
T11.1-3, Sh. 1	1	7/89
T11.1-3, Sh. 2	3	12/95
T11.1-3, Sh. 3	1	7/93
F11.1-1	Deleted	7/87
F11.1-1A	7	12/97
F11.1-1B	9	6/2000
F11.1-2	Deleted	
F11.1-2A	6	6/2000
F11.1-2B	7	6/99
F11.1-2C	5	12/97
F11.1-3	7	6/2000
11.2-1	3	6/2000
11.2-2	3	3/2001
11.2-3	3	3/2001
11.2-4	2	3/2001
11.2-5	2	3/2001
11.2-6	3	3/2001
11.2-7	5	3/2001
11.2-8	5	6/2000
11.2-9	2	7/93
11.2-10	5	6/2000
11.2-11	5	6/2000
11.2-12	5	6/2000
11.2-13	2	7/93
11.2-14	7	6/99
11.2-15	5	6/99
11.2-16	4	12/95
11.2-17	8	3/2001
11.2-18	9	3/2001
11.2-19	9	3/2001
11.2-20	6	12/95
11.2-21	4	6/2000
11.2-22	3	7/93
11.2-23	5	6/2000
11.2-24	6	6/2000
11.2-25	5	6/2000
11.2-26	4	12/97
11.2-27	5	3/2001
11.2-28	6	3/2001
11.2-29	0	3/2001
11.2-30	Deleted	7/88
11.2-31	Deleted	7/88
11.2-32	Deleted	7/88
11.2-33	Deleted	7/88
11.2-34	Deleted	7/88

Page No.	Rev No.	Date
T11.2-1	0	7/82
T11.2-2	0	7/82
T11.2-3	0	7/82
T11.2-4	0	7/82
T11.2-5	0	7/82
T11.2-6	0	7/82
T11.2-7, Sh. 1	8	6/2000
T11.2-7, Sh. 2	5	6/2000
T11.2-7A, Sh. 1	4	6/2000
T11.2-7A, Sh. 2	5	6/2000
T11.2-7A, Sh. 3	3	12/95
T11.2-7B	3	6/2000
T11.2-8	0	7/82
T11.2-9	1	7/84
T11.2-10	0	7/82
T11.2-11	0	7/82
T11.2-12	0	7/82
T11.2-13	0	7/82
T11.2-14	1	7/88
T11.2-15	1	7/88
T11.2-16	Deleted	7/88
T11.2-17	1	7/83
T11.2-18	1	7/83
T11.2-19	1	7/83
T11.2-20	1	7/83
T11.2-21	1	7/83
T11.2-22	1	7/83
T11.2-23	1	7/83
F11.2-1	1	6/99
F11.2-2	Deleted	7/90
F11.2-2A	3	6/99
F11.2-2B	2	6/99
F11.2-3	0	7/82
F11.2-4	4	7/94
F11.2-5	1	7/84
F11.2-6	1	7/88
F11.2-7	1	7/88
	Chapter 12	
12.1-1	5	6/2000
12.1-2	9	6/2000
12.1-3	10	6/2000
12.1-4	11	12/97
12.1-5	11	6/2000
12.1-6	12	6/2000
12.1-7	9	6/2000
12.1-8	9	6/2000
12.1-9	8	12/97
12.1-10	10	6/2000
12.1-11	9	3/2001
12.1-12	10	6/2000
12.1-13	8	12/97

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
12.1-14	Deleted	12/97
12.1-15	9	6/2000
12.1-16	9	6/2000
12.1-17	5	6/2000
12.1-18	5	6/2000
12.1-19	3	3/2001
12.1-20	0	12/95
F12.1-1	9	6/99
F12.1-2	9	6/2000
F12.1-3	7	6/2000
F12.1-4	8	6/2000
F12.1-5	10	6/99
F12.1-6	Deleted	12/95
F12.1-6A	3	6/2000
F12.1-6B	2	12/97
F12.1-6C	3	6/2000
F12.1-6D	2	12/97
12.2-1	6	6/99
12.2-2	Deleted	7/89
12.2-3	Deleted	7/89
12.2-4	Deleted	7/89
12.2-5	Deleted	7/89
12.2-6	Deleted	7/89
12.2-7	Deleted	7/89
12.2-8	Deleted	7/89
12.2-9	Deleted	7/89
12.2-10	Deleted	7/89
12.2-11	Deleted	7/89
12.2-12	Deleted	7/89
12.2-13	Deleted	7/89
12.2-14	Deleted	7/89
12.2-1	1	12/95
T12.3-1, Sh. 1	12	6/2000
T12.3-1, Sh. 2	6	6/2000
T12.3-1, Sh. 3	2	12/97
12.4-1	1	6/2000
12.5-1	9	6/2000
12.5-2	5	12/95
12.5-3	5	12/97
12.6-1	7	12/95
12.7-1	8	6/2000
	Chapter 13	
13.1-1	0	7/82
13.1-2	0	7/82
T13.1-1, Sh. 1	0	7/82
T13.1-1, Sh. 2	0	7/82
T13.1-1, Sh. 3	0	7/82
T13.1-1, Sh. 4	0	7/82
T13.1-1, Sh. 5	0	7/82
T13.1-1, Sh. 6	0	7/82
T13.1-1, Sh. 7	0	7/82
T13.1-1, Sh. 8	0	7/82
T13.1-1, Sh. 9	0	7/82
T13.1-1, Sh. 10	0	7/82
T13.1-1, Sh. 11	0	7/82
T13.1-1, Sh. 12	0	7/82
13.2-1	0	7/82
13.2-2	1	12/95
13.2-3	0	7/82
13.3-1	0	7/82
13.3-2	0	7/82
13.3-3	0	7/82
13.3-4	0	7/82
13.3-5	0	7/82
T13.3-1, Sh. 1	0	7/82
T13.3-1, Sh. 2	0	7/82
T13.3-1, Sh. 3	0	7/82
T13.3-1, Sh. 4	0	7/82
T13.3-1, Sh. 5	0	7/82
13.4-1	1	7/94
13.4-2	0	7/82
13.4-3	0	7/82
13.4-4	0	7/82
13.4-5	0	7/82
13.4-6	0	7/82
13.4-7	0	7/82
13.4-8	0	7/82
13.4-9	0	7/82
13.4-10	0	7/82
13.4-11	0	7/82
13.4-12	0	7/82
13.4-13	0	7/82
13.4-14	0	7/82
13.4-15	0	7/82
F13.4-1	0	7/82
F13.4-2	0	7/82
F13.4-3	0	7/82
	Chapter 14	
14.1-1	2	6/99
14.1-2	4	6/99
14.1-3	7	3/2001

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date	Page No.	Rev No.	Date
14.1-4	6	6/99	14.1-56	7	6/99
14.1-5	5	6/99	14.1-57	2	6/99
14.1-6	4	6/99	14.1-58	3	6/99
14.1-7	4	6/99	14.1-59	2	6/99
14.1-8	6	6/99	14.1-60	3	6/2000
14.1-9	6	6/99	14.1-61	3	6/2000
14.1-10	5	6/99	14.1-62	2	6/99
14.1-11	4	6/99	14.1-63	1	6/99
14.1-12	5	6/99	14.1-64	2	6/99
14.1-13	4	6/99	14.1-65	1	6/99
14.1-14	6	6/99	14.1-66	1	6/99
14.1-15	5	6/99	14.1-67	1	6/99
14.1-16	5	6/99	14.1-68	1	6/99
14.1-17	5	6/99	14.1-69	1	6/99
14.1-18	5	6/99	14.1-70	1	6/99
14.1-19	5	6/99	14.1-71	1	6/99
14.1-20	5	6/99	14.1-72	2	6/2000
14.1-21	5	6/99	14.1-73	1	6/99
14.1-22	5	6/99	14.1-74	1	6/99
14.1-23	5	6/99	14.1-75	2	6/99
14.1-24	5	6/99	14.1-76	3	6/2000
14.1-25	9	6/2000	14.1-77	2	6/2000
14.1-26	7	6/99	14.1-78	2	6/99
14.1-27	6	6/99	14.1-79	1	6/99
14.1-28	6	6/99	14.1-80	1	6/99
14.1-29	6	6/99	14.1-81	2	6/99
14.1-30	7	6/99	14.1-82	Deleted	6/99
14.1-31	7	6/2000	14.1-83	Deleted	6/99
14.1-32	6	6/99	14.1-84	Deleted	6/99
14.1-33	7	6/99	14.1-85	Deleted	6/99
14.1-34	6	6/99	14.1-86	Deleted	6/99
14.1-35	6	6/99	14.1-87	Deleted	6/99
14.1-36	6	6/99	14.1-88	Deleted	6/99
14.1-37	7	3/2001	14.1-89	Deleted	6/99
14.1-38	6	6/99	14.1-90	Deleted	6/99
14.1-39	7	6/99	14.1-91	Deleted	6/99
14.1-40	5	6/99	14.1-92	Deleted	6/99
14.1-41	5	6/99	14.1-93	Deleted	6/99
14.1-42	6	6/99	14.1-94	Deleted	6/99
14.1-43	5	6/99	T14.1-0	0	6/99
14.1-44	5	6/99	T14.1-1	1	6/99
14.1-45	5	6/99	T14.1-2	1	6/99
14.1-46	5	6/99	T14.1-3	1	6/99
14.1-47	5	6/99	T14.1-4	1	6/99
14.1-48	5	6/99	T14.1-5	1	6/99
14.1-49	5	6/99	T14.1-6	1	6/99
14.1-50	6	6/99	T14.1-7	1	6/99
14.1-51	5	6/99	T14.1-8	1	6/99
14.1-52	5	6/99	T14.1-9	2	6/2000
14.1-53	6	6/2000	T14.1-10, Sh. 1	1	6/99
14.1-54	5	6/99	T14.1-10, Sh. 2	0	6/99
14.1-55	5	6/99	T14.1-11, Sh. 1	1	6/99

STATUS OF FSAR PAGES, TABLES and FIGURES

Page No.	Rev. No.	Date
	Chapter 17	
17.1-1	0	7/82
17.2-1	7	6/2000
17.2-2	7	6/2000
17.2-3	9	6/2000
17.2-4	9	6/2000
17.2-5	7	6/99
17.2-6	5	6/99
17.2-7	5	6/2000
17.2-8	5	12/95
17.2-9	4	12/95
17.2-10	5	6/2000
17.2-11	5	12/97
17.2-12	5	12/95
17.2-13	6	3/2001
17.2-14	5	3/2001
17.2-15	5	3/2001
17.2-16	5	3/2001
17.2-17	5	3/2001
17.2-18	4	3/2001
17.2-19	5	3/2001
17.2-20	6	3/2001
17.2-21	4	3/2001
17.2-22	5	3/2001
17.2-23	4	3/2001
17.2-24	4	3/2001
17.2-25	4	3/2001
17.2-26	4	3/2001
17.2-27	6	3/2001
17.2-28	7	3/2001
17.2-29	6	3/2001
17.2-30	0	3/2001
17.2-31	Deleted	7/95
17.2-32	Deleted	7/95
F17.2-1	12	6/2000
F17.2-2	12	6/2000
F17.2-3	10	6/2000
F17.2-4	12	6/2000
F17.2-5	7	6/99
17.2A-1	1	7/84
17.2B-1	5	6/99
17.2B-2	0	7/82
17.2B-3	0	7/82
17.2B-4	2	12/95
17.2B-5	1	6/2000
17.2B-6	0	6/99
17.2C-1	12	6/2000
17.2C-2	5	12/97
17.2C-3	0	12/97
17.2D-1	2	7/84
17.2D-2	8	6/2000
17.2E-1	1	3/2001

Page No.	Rev No.	Date
17.2E-2	1	3/2001
17.2E-3	0	3/2001
17.2E-4	0	3/2001
17.2E-5	0	3/2001
17.2F.1	0	3/2001
17.2F-2	0	3/2001

Ch. 1

effects of the natural phenomena and (3) the importance of the safety functions to be performed.

Seismic Class I structures were designed to maintain within the allowable stress limit a combination of primary steady state stresses with seismic stresses resulting from the application of seismic motion with a maximum ground acceleration of 0.05 g acting in the vertical and 0.1 g acting in the horizontal planes simultaneously. Also, primary steady state stresses when combined with seismic stresses resulting from the application of seismic motion with a maximum ground acceleration of 0.10 g acting in the vertical and 0.15 g acting in the horizontal planes simultaneously, were limited so that the function of the component system or structure shall not be impaired as to prevent a safe and orderly shutdown of the plant.

The plant is safeguarded from tornadoes by the combined use of buildings and structures, designed to withstand tornadoes, and redundancy of components. All seismic Class I buildings and structures were designed to withstand tornado winds corresponding to 300 mph tangential velocities, traverse velocities of 60 mph and a differential pressure drop of 3 psi in 3 seconds with no loss of function. The exceptions to this include areas without safety related equipment or redundant equipment as discussed in FSAR Section 16.2-2

Furthermore, using a Probable Maximum Hurricane at the Battery of 130 mph and an inland reduction factor of 0.7, a wind speed of 90 mph due to hurricane was derived for use in the design at Indian Point.

From the evaluation of flooding conditions at Indian Point, done by Environmental Science and Engineering Consultants (Section 2.5), it was concluded that the maximum elevation of water at Indian Point due to flooding and wave runup is 15 feet or less. The consultants arrived at the above conclusion after assuming a critical set of simultaneous occurrences of the following three severe conditions:

1) Probable maximum precipitation over the Ashokan Reservoir resulting in a dam failure

2) Runoff generated by standard project precipitation over the Hudson Basin

3) Peak storm surge resulting from standard project hurricane for the New York Harbor area.

Although the simultaneous occurrence of the above three conditions is extremely remote, using the above conditions or various other conditions described in Chapter 2, the flood effects were not governing in the design of the plant. The combination of the elevation of the plant structures, the load-bearing capacity of the intake structure and the Technical Requirements Manual (TRM) requirements on plant operation and service water pump protection, result in acceptable conditions to protect the plant against flooding.

From the total group of meteorological conditions using a wind speed of 30 mph, an effective fetch length of 5.2 miles and an average water depth of 25 feet, the significant wave height at the site will be 2.8 feet with a significant wave period of 5.9 sec. Maximum wave height will be 5.0 feet. (Section 2.6)

During a probable maximum hurricane condition with a wind speed of 90 mph, a water depth of 35 feet, and a fetch of 5.2 miles the wave height at the site will be 9.1 feet. (Section 2.6)

From the above, it is evident that design loads due to the effects of tornadoes, hurricanes and flood were determined after considering the most severe of the natural phenomena that have been historically reported for the site and surrounding area. It is also evident that sufficient margin for limited accuracy, quantity, and period of time in which historical data have been accumulated was appropriate for each phenomenon.

The earthquake response spectra were developed from the average acceleration velocity displacement curves presented in TID-7024, Nuclear Reactors and Earthquake, for large-magnitude earthquakes at moderate distances from the epicenter. As such, the curves are made up of the combined normalized response spectrum determined from components of four strong-motion ground accelerations: El Centro, California, December 30, 1934; El Centro, California, May 18, 1940; Olympia, Washington, April 13, 1949; and Taft, California, July 21, 1952.

In investigating the overall and local structural effects, the following appropriate combinations of the effects of normal and accident conditions with effects of the natural phenomena were made:

The maximum tornado wind load was combined with missile load, dead load and live load, or with 3 psi negative pressure, and missile loads yielding the most conservative load combination and/or the highest stress condition.

Tornado loads act independently of other severe loads, and were found to be small by comparison to seismic loading.

With the exception of the Containment, all other seismic Class I structures used the following load combinations:

1) Primary steady state stresses were combined with the seismic stress resulting from the application of seismic motion with a maximum ground acceleration of 0.05g acting in the vertical and 0.1 g acting in the horizontal planes simultaneously. Under this combination the stresses were maintained within the allowable stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards; e.g., ASME Boiler and Pressure Vessel Code, USAS B31.1, "Code for Pressure Piping," ACI 318 "Building Code Requirements for Reinforced Concrete," and

1.3.2 Protection by Multiple Fission Product Barriers (Criteria 10 to 19)

Reactor Design (Criterion 10)

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

The reactor core, with its related control and protection system, was designed to function throughout its design lifetime without exceeding acceptable fuel damage limits. The core design, together with reliable process and decay heat removal systems, provides for this capability under all expected conditions of normal operation with appropriate margins for uncertainties and anticipated transient situations, including the effects of the loss of reactor coolant flow, trip of the turbine generator, loss of normal feedwater and loss of all offsite power.

The Reactor Control and Protection System was designed to actuate a reactor trip for any anticipated combination of plant conditions, when necessary, to ensure a minimum Departure from Nucleate Boiling (DNB) ratio equal to or greater than the applicable limit for the analyzed accidents.

The integrity of fuel cladding is ensured by preventing excessive clad heating, excessive cladding stress and strain. This was achieved by designing the fuel rods so that the following conservative limits are not exceeded during normal operation or any anticipated transient condition:

1) Minimum DNB ratio equal to or greater than the applicable limit;

2) Fuel center temperature below melting point of UO_2;

3) Internal gas pressure less than the nominal external pressure (2250 psia) even at the end of life;

4) Clad stresses less than the Zircaloy and ZIRLO™ yield strengths;

5) Clad strain less than 1%;

6) Cumulative strain fatigue cycles less than 80% of design strain fatigue life.

The ability of the fuel, when designed and operated according to these criteria, to withstand postulated normal and abnormal service conditions as shown by analyses described in Chapter 14 has been ensured. These analyses have been and will be amended as necessary for each cycle by the corresponding reload safety evaluations.

The Indian Point 3 Technical Specifications establish reactor criticality limits on moderator temperature coefficient and minimum temperature.

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.

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 was taken to calculate the power coefficient, based on operating experience of existing Westinghouse cores. Section 3.2 provides 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.

Reactor Inherent Protection (Criterion 11)

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

Prompt compensatory reactivity feedback effects are assured when the reactor is critical by the negative fuel temperature effect (Doppler effect) and by the non-positive operational limit on the moderator temperature coefficient of reactivity. The negative Doppler coefficient of reactivity is assured by the use of low enrichment fuel, the non-positive moderator temperature coefficient of reactivity is assured by keeping the dissolved absorber concentration below a certain limit through the use of burnable poison.

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, was evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients were required to couple the response of the core neutron multiplication to the variables which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range

The NDTT shift of the vessel material and welds, due to radiation damage effects is monitored by a radiation damage surveillance program which conforms with ASTM E-185 standards.

Reactor vessel design was based on the transition temperature method of evaluating the possibility of brittle fracture of the vessel material as a result of operations such as leak testing and plant heatup and cooldown.

To establish the service life of the Reactor Coolant System components as required by the ASME (Section III) Boiler and Pressure Vessel Code for Class "A" vessels, the unit operating conditions have been established for the 40 year design life. These operating conditions include the cyclic application of pressure loadings and thermal transients.

The Reactor Coolant System is classified as Class I for seismic design, requiring that there will be no loss of function of such equipment in the event of the assumed maximum potential ground acceleration acting in the horizontal and vertical directions simultaneously, when combined with the primary steady state stresses. (Section 4.1.4)

The heatup and cooldown curves for the plant are based on the actual measured fracture toughness properties of the vessel materials. Maximum allowable pressures as a function of the rate of temperature change and allowable pressures as a function of the rate of temperature change, and the actual temperature, are established according to the methods given in Appendix G, Protection Against Non-Ductile Failure, published in the 1972 Summer Addenda of Section III the ASME Pressure Vessel and Boiler Code. The original RCS heatup and cooldown curves for up to 9.26 effective full power years (EFPYs) of reactor operation were developed from the analysis of capsule T. Subsequent analyses of capsules Y and Z did not require that changes be made to these curves until a new methodology to predict the effect of neutron radiation on reactor vessel materials was presented by Generic Letter 88-11. Consequently, new heatup and cooldown curves for 13.3 EFPYs based on the analysis of capsule Z in accordance with the methodology of Regulatory Guide 1.99, Revision 2, "Radiation Embrittlement of Reactor Vessel Materials." With increasing years of service, these curves were revised for 11 and 13.3 EFPYs. These curves are given in the IP3 Technical Specifications. (Further details are given in Chapter 4.)

Inspection of Reactor Coolant Pressure Boundary (Criterion 32)

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

Post-operational examinations as set forth in ASME Section XI are performed to the fullest extent practical at the required intervals.

The structural integrity of the Reactor Coolant System is maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence, as a result of the inspections listed in Table 4.2-1 of the Technical Specifications, that potential defect implications have initiated or enlarged shall be investigated, including evaluation of comparable areas of Reactor Coolant System.

Nondestructive test methods, personnel, equipment and records conform to the requirements of ASME B&PV Code, Section XI.

The use of conventional nondestructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several steps were incorporated into the design and manufacturing procedures in preparation for nondestructive test techniques. These are:

1) Shop ultrasonic examinations were performed on all thermally clad surfaces to an acceptance and repair standard to assure an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed is ¼" x ¾".

2) The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit future positioning of test equipment without obstruction.

3) Selected areas of the reactor vessel were ultrasonic tested and mapped during the manufacturing stage to facilitate the in-service inspection program by establishing baselines for later testing. The areas selected for ultrasonic testing mapping are:

   a) Vessel flange radius, including the vessel flange to upper shell weld

   b) Middle shell course

   c) Lower shell course above the radial core supports

   d) Exterior surface of the closure head from the flange knuckle to the cooling shroud

   e) Nozzle to upper shell weld

   f) Middle shell to lower shell weld

   g) Upper shell to middle shell weld

The preoperational ultrasonic testing of these areas was performed after hydrostatic testing of the reactor vessel.

Pre-operational performance tests of the components were performed in the manufacturer's shop. The pressure containing parts of the pump were hydrostatically tested in accordance with Paragraph UG-99 of Section VIII of the ASME Code. Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head and at additional points to verify performance characteristics. NPSH was established at design flow by means of adjusting suction pressure for a representative pump. This test was witnessed by qualified Westinghouse personnel.

The remote operated valves in the Safety Injection System are motor operated. Shop tests for each valve included a hydrostatic pressure test, leakage tests, a check of opening and closing time, and verification of torque switch and limit switch settings. The ability of the motor operator to move the valve with the design differential pressure across the gate was demonstrated by opening the valve with an appropriate hydrostatic pressure on one side of the valve.

The recirculation piping and accumulators were hydrostatically tested at 150 percent of design pressure.

The service water and component cooling water pumps were thoroughly tested prior to initial operation.

Periodic testing of the Safety Injection System components and all necessary support systems at power is a portion of the Authority's test program. The safety injection and residual heat removal pumps are to be tested in accordance with the Indian Point 3 Inservice Testing Program, to check the operation of the starting circuits, verify the pumps are in satisfactory running order, and verification is made that required discharge head is attained. No inflow to the Reactor Coolant System will occur whenever the reactor coolant pressure is above 1500 psi. If such testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions defined in the Technical Specifications. These conditions include the period within which the component should be restored to service and the capability of the remaining equipment to meet safety limits within such a period.

The operation of the remote stop valves in the accumulator tank discharge line may be tested by opening the remote test valves just downstream of the stop valve. Flow through the test line can be observed on instruments and the opening and closing of the discharge line stop valves can be sensed on this instrumentation. Test circuits are provided to periodically examine the leakage back through the check valves and to ascertain that these valves seat whenever the reactor system pressure is raised.

This test is routinely performed when the reactor is being returned to power after an outage and the reactor pressure is raised above the accumulator pressure. If leakage through a check valve should become

excessive, the isolation valve would be closed. (The safety injection actuation signal will cause this valve to open should it be in the closed position at the time of a Loss-of-Coolant Accident.) The performance of the check valves has been carefully studied and it is highly unlikely that the accumulator lines would have to be closed because of leakage.

The recirculation pumps are normally in a dry sump. Minimum flow testing of these pumps is performed during refueling operation by filling the recirculation sump and opening the miniflow valve on the discharge of the pump and directing the flow back to the sump. Those service water and component cooling pumps which are not running during normal operation may be tested by alternating them with the operating pumps.

The content of the accumulators and the Refueling Water Storage Tank are sampled periodically to assure that the required boron concentration is present.

System testing is conducted during plant shutdown to demonstrate proper automatic operation of the Safety Injection System. An actual or simulated safety injection test signal is applied to initiate automatic action and verification is made that the components receive the safety injection signal in the proper sequence. The Safety Injection and Residual Heat Removal pumps are blocked from starting. Isolation valves in the injection lines are blocked closed so that flow is not introduced into the Reactor Coolant System. The system test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry. The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly. A complete system test cannot be performed when the reactor is operating because a safety injection signal would cause a reactor trip. The method of assuring complete operability of the Safety Injection System is to combine the system test performed during plant shutdown with more frequent component tests, which can be performed during reactor operation.

The accumulators and the injection piping up to the final isolation valve are maintained full of borated water while the plant is in operation. The accumulators and the high head injection lines are refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection lines. A small test line is provided for this purpose in each injection header.

Flow in each of the safety injection headers and in the main flow line for the residual heat removal pumps is monitored by a local flow indicator. Pressure instrumentation is also provided for the main flow paths of the safety injection and residual heat removal pumps. Accumulator isolation valves are blocked closed for this test.

The eight-switch sequence for recirculation operation is tested following the above injection phase test to demonstrate proper sequencing of valves and pumps. The recirculation pumps are blocked from start during this test.

The external recirculation flow paths are hydrotested during periodic retests at the operating pressures. This is accomplished by running each pump which could be utilized during external recirculation (safety injection and residual heat removal pumps) in turn at near shutoff head

system redundancy and independence are provided such that no single active or passive component failure can negate the minimum requirements of operation.

Inspection of Containment Atmosphere Cleanup Systems (Criterion 42)

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

Access is available for visual inspection of the Containment Air Recirculation Cooling and Filtration System components including fans, cooling coils, dampers, filter units and ductwork. Provisions were made for ready removal of the filters for inspections and testing. (Section 6.4)

Design provisions were made to the extent practicable to facilitate access for periodic visual inspections of all important components of the Containment Spray System.

Periodic operating checks of the Hydrogen Recombiner System are specified in Technical Specification Section 3.6.8 and described in the FSAR. The portion of the system located outside the Containment should be readily accessible for inspection at any time, while the Containment portion would be accessible for inspection during reactor shutdown.

Testing of Containment Atmosphere Cleanup Systems (Criterion 43)

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

The Containment Air Recirculation Cooling and Filtration system was designed to the extent practicable so that components can be tested periodically and after maintenance for operability and functional performance. The air recirculation and cooling units are in operation on an essentially continuous schedule during plant operation and no additional periodic tests are required. (Section 6.4)

The closed dampers, which allow for charcoal filter bypass during normal operation, can be periodically tested in a non-operating unit by activating controls and verifying deflection by instruments in the Control Room. Representative sample elements in each of the activated carbon filter plenums are removed periodically during shutdowns and tested to verify their continued efficiency. After reinstallation, the filter assemblies are tested in place in accordance with the Technical Specifications to determine integrity of the flow path.

Means were provided to test initially under conditions as close to design as was practicable the full operational sequence that would bring the system into action, including transfer to the emergency diesel generator power source. Surveillance testing of the containment air filtration system is covered in the Technical Specifications.

One of the design bases for the Hydrogen Recombination System was that the system shall be testable during normal operating conditions of the plant. Operating checks of the system are performed periodically, as required by the Technical Specifications and as discussed in the FSAR.

Testing of the Containment Spray System is addressed in the discussion for Criterion 40.

Cooling Water (Criterion 44)

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

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

A closed-loop Component Cooling Water System and a once-through Service Water System are provided to transfer heat loads from structures, systems and components important to safety to an ultimate heat sink. The component cooling system transfers heat loads to the Service Water System via component cooling heat exchangers. The Service Water System takes water from the Hudson River and supplies cooling water for the Component Cooling Water

Strength Design" of ACI 318-63. Because of the refinement of the analysis and the restrictions on construction procedures, the load factors in the design primarily provide for a safety margin on the load assumptions. Specific combined load equations used in design are presented in Appendix 5A.

All structural components were designed to have a capacity required by the most severe loading combination.

The design included consideration of both primary and secondary stresses. The load capacity in structural members was based on the ultimate strength values presented in Part IV-B of ACI-318 as reduced by the capacity reduction factor "ϕ" which provides for the possibility that small adverse variations in material strengths, workmanship, dimensions, and control, while individually within required tolerances and the limits of good practice, occasionally may combine to result in undercapacity.

For the liner steel the factor "ϕ" is 0.95 for tension. For compression and shear, the primary membrane liner stress was maintained below 0.95 yield and elastic stability has been assured as a function of liner anchorage requirements.

The liner was designed to assure that no strains greater than the strain at the guaranteed yield point will occur at the factored loads except in regions of local stress concentrations or stresses due to secondary load effects in which case the liner strain is limited to 0.5 percent. Sufficient anchorage is provided to assure elastic stability of the liner. The basic design concept utilizing stud anchorage of the liner plate to the concrete structure assures stud failure due to shear, tension or bending stress without the stud connection causing failure or tear of the liner plate. The studs in the 0.5-inch plate were installed on a 24" horizontal and 28" vertical grid and, in the 0.375 inch plate, on a 24" horizontal and 14" vertical grid. The design considered the possibility of daily stress reversals due to ambient temperature changes for the life of the plant and the fatigue limit of the studs exceeds the design requirements. (Section 5.1.2)

Service temperatures during operation, maintenance and testing are less severe than those accompanying the containment design basis conditions and therefore will not induce brittle fracture of the containment liner.

<u>Capability for Containment Leakage Rate Testing (Criterion 52)</u>

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

After completion of the containment structure and installation of all penetrations and weld channels, integrated leakage rate tests were performed

prior to initial plant operations at a reduced test pressure and at the calculated peak accident pressure to establish the respective measured leakage rates and to verify that the leakage rate at the peak accident conditions was no greater than 0.075 percent by weight per day of the containment steam-air atmosphere at the calculated peak accident conditions. Leak rate testing of the containment is performed in accordance with Technical Specification 5.5.15. The leakage rate program is in accordance with the guidance contained in Regulatory Guide 1.163, except as noted in the Technical Specifications.

The peak accident pressure integrated leakage rate test is conducted at periodic intervals during the life of the plant, and also as appropriate in the event of major maintenance or major plant modifications. A leak rate test at the peak accident pressure using the same test method as the initial leak rate test can be performed at any time during the operational life of the plant, provided the plant is not in operation and precautions are taken to protect instruments and equipment from damage. (Section 5.1.7)

Penetrations were designed with double seals, which are continuously pressurized above accident pressure. The large access openings, such as the Equipment Hatch and Personnel Air Locks, are equipped with double gasketed doors and flanges with the space between the gaskets connected to the pressurization system. The system utilizes a supply of clean, dry compressed air, which places the penetrations under an internal pressure above the peak calculated accident pressure.

A permanently piped monitoring system continuously measures leakage from all penetrations.

Leakage from the monitoring system is checked by continuous measurement of the integrated makeup air flow. In the event excessive leakage is discovered, each penetration can then be checked separately at any time.

Capability is provided to the extent practical for testing the functional operability of valves and associated apparatus during periods of reactor shutdown.

Initiation of containment isolation employs coincidence circuits which allow checking of the operability and calibration of one channel at a time.

<u>Provisions for Containment Testing and Inspection (Criterion 53)</u>

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

To provide for testing the larger penetrations, branch pressurizing lines are installed from one of the zones to:

a) The double-gasketed space on each hatch of the Personnel Air Lock

b) The double-gasketed space at the Equipment Hatch Flange

c) The pressurized zones in the spent fuel transfer tube

d) The spaces between the two butterfly valves in the purge supply and exhaust ducts

e) The two spaces between the three butterfly valves in the containment pressure relief line

f) The spaces between double containment isolation valves in the steam jet air ejector return line to containment and in the containment radiation monitor inlet outlet lines

The makeup air flow to the penetrations and liner weld joint channels during normal operation is only an indication of the potential leakage from the containment. It does indicate the leakage from the pressurization system, and degree of accuracy is increased when correlated with the results of the full scale containment leak rate tests. The criteria for selection of operating limits for air consumption of the pressurization system are based upon the integrated containment leak rate test acceptance criterion and upon the maintenance of suitable reserve air supplies in the static reserves consisting of the air receivers and nitrogen cylinders. A summary of these operating limits is as follows:

1) A baseline air consumption rate was established for each of the four pressurization headers at the time of successful completion of the preoperational integrated containment leak rate tests. Unexplained increases from the consumption rate require routine investigation and location of the point of leakage.

2) The upper limit for long-term air consumption for the pressurization system is 0.2% of the containment volume per day (sum of four headers) at the system operating pressure, contingent on the following:

    a) Pressure in all pressurization zones is maintained above incident pressure

    b) Air supply is maintained from the compressed air systems with compressors running

c) The full complement of air receivers (4) and standby nitrogen cylinders (3) are charged. This is consistent with maintenance of a 24 hour supply.

A variable area flow sensing device is located in each of the headers supplying makeup air to the four pressurization zones. Signal output from each of the four flow sensors is applied to an integrating recorder located in the control room. High flow alarms are available to alert the operator in the control room. The sensitivity of the flowmeters is well within the maximum leakage of the pressurization system.

<u>Piping Systems Penetrating Containment (Criterion 54)</u>

Criterion: Piping systems penetrating primary reactor containment shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems.

Such piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Piping systems which penetrate the Reactor Containment have isolation capabilities commensurate with the systems' design in accomplishing safety related functions or providing isolation of the containment from the outside environment under postulated accident conditions. Piping penetrating the Containment was designed for pressures at least equal to the containment design pressure and, as a minimum, those portions of piping systems which are essential to the isolation function are capable of withstanding the maximum potential seismic loads.

For those systems which require isolation, redundant barriers are utilized to ensure that the failure of one valve to close will not prevent isolation of the penetration. The containment isolation provisions are discussed in detail in Section 5.2.

Isolation valves which are located in lines connecting to the Reactor Coolant System or which could be exposed to the containment atmosphere under postulated accident conditions are sealed by an Isolation Valve Seal Water System which injects water or gas at a pressure slightly higher than the containment design pressure between the isolation barriers. Containment penetrations and welds are sealed by the Containment Penetration and Weld Channel Pressurization System. In addition to providing seals on penetration isolation barriers, these systems may be utilized for leakage detection. The design and operation of the seal system are described in Sections 6.5 and 6.6. Additional leakage detection provisions are discussed in Section 6.7.

Leakage rate testing of the containment is performed in accordance with Technical Specification 5.5.15. The leakage rate program is in accordance with the guidance contained in Regulatory Guide 1.163, except as noted in the Technical Specifications.

Provisions for leakage testing are discussed in Section 5.1.7 and 5.2.

Reactor Coolant Pressure Boundary Penetrating Containment (Criterion 55)

Criterion: Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

3) One locked closed isolation valve inside and one automatic isolation valve outside of containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

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

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

Lines which penetrate the Containment and connect to the Reactor Coolant Pressure Boundary were designed with isolation capabilities sufficient to preclude the release of significant amounts of radioactivity. The containment isolation provisions of each line are consistent with the function of

the line during normal operation and safety function of the line during and after the Design Basis Accident.

Effluent lines, except the residual heat removal cooldown line, which connect to the Reactor-Coolant Pressure Boundary and which are normally or intermittently open during reactor operation, but which are not required for safe shutdown, are provided with at least two automatic isolation or two closed manual isolation valves in series located outside of containment. Automatic seal water injection is provided for these valves, as described in Sections 5.2.2. and 6.5.2.

A single, normally closed, manually operated, double disc gate valve is provided on the residual heat removal cooldown line outside of containment for isolation purposes. Because of pressure considerations described in Section 5.2.2, the valve can be sealed between the discs by nitrogen from the Isolation Valve Seal Water System. The seal is manually initiated, as required.

Effluent and influent lines connected to the Reactor Coolant Pressure Boundary which are part of systems with essential safety functions and, therefore, must function during or after the Design Basis Accident are provided with containment isolation capabilities commensurate with the required function of the system which the line serves.

Lines in the above category, which are connected to closed systems outside of containment, utilize a single remote manually motor operated double disc gate valve outside of containment as the first isolation barrier. The closed system functions as the second barrier. Manually initiated sealing systems are available to pressurize between the discs of the gate valve after use of the penetration is terminated post accident.

Lines associated with the Residual Heat Removal System which do not terminate in a closed system were designed with isolation provisions which reflect the required function of the line. (Section 5.2)

The residual heat removal return line isolation provisions consist of a check valve inside containment and outside of the missile barrier as the inboard isolation barrier and a normally open, remote manual motor operated double disc gate valve located outside of containment as the outboard isolation barrier. A manually initiated nitrogen seal is available to the outboard valve.

Two normally closed isolation gate valves in series are provided outside of containment for the recirculation pump discharge sample line. The lines between the isolation valve may be sealed by manually initiating a nitrogen seal from the Isolation Valve Seal Water System, as described in Section 5.2.2.

Since the minimum flow line for the residual heat removal pumps must be open upon pump start, two normally open motor operated valves are provided

Adequate shielding for radiation protection is provided during reactor refueling by conducting all spent fuel transfer and storage operations underwater. This permits visual control of the operation at all times while maintaining low radiation levels, less than 2.0 mr/hr, for periodic occupancy of the area by operating personnel. Pit water level is indicated, and water removed from the pit must be pumped out since there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

Gamma radiation is continuously monitored in the auxiliary building. A high level signal is alarmed locally and is annunciated in the Control Room.

Auxiliary shielding for the Waste Disposal System and its storage components was designed to limit the dose rate to levels not exceeding 0.75 mr/hr in normally occupied areas, to levels not exceeding 2.0 mr/hr in intermittently occupied areas and to levels not exceeding 15 mr/hr in limited occupancy areas. (Section 11.2)

A controlled leakage building designed for a negative pressure of 0.50 inches of water minimum, permanently encloses the fuel pool. The design features of the fuel handling building that provide this leaktightness include the following items:

1) Special sealing features at joints that include:

   a) Sealing off all edges and ends of the walls with a combination of caulking and relatively soft neoprene strip,

   b) Installation of necessary additional closure flashings at the extremities and at openings,

   c) Supplying additional caulking in vertical and horizontal joints of liner panels,

   d) Furnishing liner panels in sufficient thickness to seat well on girt spacings and resist flexing in addition to withstanding the normal design loads, and

   e) Providing additional fastening for liner panels.

2) Personnel and rolling steel truck doors with inflatable air seals. These seals are inflated upon a high radiation alarm from R-5, although R-5 operability does not require this function.

3) Motor operated dampers designed to fail closed are installed on the discharge side of the two supply fans.

Prior to handling operations when irradiated fuel is within the fuel handling building, tests were performed to verify the building leak tightness. Fuel handling operations in the Fuel Storage Building are detailed in Section 3.7.13 of Technical Specifications and Section 9.5 of the FSAR.

Carbon filters, together with suitable face dampers and manual isolation devices are part of the normal Fuel Storage Building Ventilation System and operate as follows:

1) The carbon filters and motor operated dampers are located in the fan plenum downsteam from the roughing and HEPA filters.

2) The fuel storage building supply air fans are provided with motor operated dampers located on the discharge side of the fan. These dampers are interlocked with their respective fan motors and arranged to close when fan motor stops and open during fan motor operation.

3) Manual isolation devices will be installed during all fuel handling operations and leak tested to ensure that all of the air from the fuel storage building is discharged through the roughing HEPA and charcoal filters.

4) A radiation indicator located in the spent fuel pit area automatically initiates the emergency mode of operation by:

   a) Stopping fuel storage building supply fans, thereby closing their respective dampers.

   b) Opening carbon filter face dampers.

5) The exhaust system has a capacity of approximately 20,000 cfm which maintains a negative pressure in the fuel storage building.

The probability of inadvertently draining the water from the cooling loop of the spent fuel pit is exceedingly low. The only means is through actions such as opening a valve on the cooling line and leaving it open when pump is operating. In the unlikely event of the cooling loop of the spent fuel pit being drained, the spent fuel storage pit itself cannot be drained and no spent fuel is uncovered since the spent fuel pit cooling connections enter near the top of the pit. With no heat removal, the time for the spent fuel pit water to rise from 120 ^{0} F to 180 ^{0} F with 76 fuel assemblies stored in the pit is approximately 13.8 hours. The temperature and level indicators in the spent fuel pit would warn the operator of the loss of cooling. This slow heatup rate of the spent fuel pit would allow sufficient time to take any necessary action to provide adequate cooling while the cooling capability of the spent fuel pit cooling loop is being restored.

Assuming that the reactor has recently been refueled and 76 assemblies are stored in the pool, and 76 assemblies were placed in the spent fuel pit, a fission product decay period of approximately 47.5 days would be required after the spent fuel was placed in the pit before the natural heat loss from the pit would be equivalent to the decay heat.

The most serious failure of this loop is complete loss of water in the storage pool. To protect against this possibility, the spent fuel storage pool cooling connections enter near the water level so that the pool cannot be either gravity-drained or inadvertently drained. For this same reason, care is also exercised in the design and installation of the fuel transfer tube. The water in the spent fuel pit below the cooling loop connections could be removed by using a portable pump.

Loss of water in the spent fuel pit and the resultant uncovering of the spent fuel by way of drains, and permanently connected systems cannot take place for the following reasons:

a) The suction of the Spent Fuel Pit Pump is taken from a point approximately six (6) feet below the surface of the pool; therefore, this pump cannot be used to uncover fuel, even accidentally

b) The Spent Fuel Pump discharges into the pool approximately seven (7) feet above the top of the spent fuel assemblies; therefore, this pipe could not accidentally become a syphon to uncover the fuel.

c) The skimmer pump takes suction from, and discharges to the surface of the pool; therefore, it could not accidentally or otherwise uncover the spent fuel

d) There are no drains on the bottom or side walls of the spent fuel; draining has to be done deliberately by a temporary pump

e) The spent fuel pit cooling loop was designed to seismic Class II, the cleanup equipment and skimmer loops were designed to seismic Class III criteria; however, their failure could not result in the uncovering of the spent fuel, as explained above

The primary source of makeup water to the spent fuel pit is the Primary Makeup Water Storage Tank, which is a seismic Class I component. The pumps and most of the piping associated with this system are also seismic Class I. The makeup water loop to the Spent Fuel Pit is seismic Class II, as is the spent fuel pit cooling loop. The cleanup equipment and skimmer loops are seismic Class III. Redundant sources of makeup water to the spent fuel pit are from the refueling water storage tank and the city water supply. In addition, there are provisions for the connection of a temporary cooling system. See Section 9.5 for further discussion of possible loss of water from the spent fuel pit and makeup capabilities.

In addition, a second Spent Fuel Pool Cooling System pump was installed identical to and in parallel with the original pump to provide installed standby pumping capability to the Spent Fuel Pool Cooling System.

The reliability of pumping capability is further enhanced by powering the two pumps from different electrical power buses. Associated piping and

valves were designed and analyzed to requirements consistent with the existing system. The modified portions of the system were as a minimum, equal to the standards of the original system and in most cases represents an upgrading of design, material, fabrication, testing and/or quality assurance.

Prevention of Criticality in Fuel Storage and Handling (Criterion 62)

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

The Indian Point 3 spent fuel storage pool is equipped with seismic Class I maximum density spent fuel storage racks for an expanded total storage capacity of 1345 fuel assemblies. A stainless steel liner insures against a loss of water. Detailed instructions, licensing bases including Technical Specification requirements and the design of the fuel handling equipment incorporating interlocks and safety features, provide assurance that no incident could occur during refueling, fuel handling, or storage operations that could result in a hazard to the public health and safety. (Section 9.5)

Borated water is used to fill the spent fuel storage pit to a concentration matching that of the reactor cavity and refueling canal during refueling operations. A shutdown margin of 5% Dk/k is maintained, in the cold condition, with all rods inserted. Periodic checks of refueling water boron concentration and the residual heat removal pump operation insure the proper shutdown margin. Direct communications between the control room operator and the manipulator operator allows immediate notification of any impending unsafe condition detected during fuel movement.

The racks are arranged and categorized in two regions based on fuel assembly enrichment and burn-up. All storage cells are bounded on four sides by boral poison sheets, except on the periphery of the pool rack array. The rack are designed to assure that a keff of less than or equal to 0.95 is maintained provided that Technical Specifications dictating placement of fuel in the spent fuel pit are followed.

The storage rack design is a free-standing welded honeycomb array of stainless steel boxes which has no grid frame structure. The racks are supported and leveled on four screw pedestals which bear directly on the pool floor. The racks are free to move horizontally, and strong hydrodynamic coupling between racks causes the racks to move together without rack-to-rack impact. The free-standing design allows any single or combination of racks to withstand a design basis seismic event without toppling or causing damage to fuel assemblies inserted within them.

The core subcritical neutron flux is continuously monitored by two source range neutron monitors, each with continuous visual indication in the

Control Room and with one audible indication in the Containment whenever core geometry is being changed.

Fuel Handling System cranes are dead-load tested before fuel movement begins. The test load assumed by the hoists or cranes must be equal to or greater than the maximum load assumed during the refueling operation. Additionally, a thorough visual inspection is made following the dead-load test and prior to fuel handling. A test of interlocks is also performed each refueling, prior t movement of core components. As excess weight interlock is provided to prevent movement of more than one fuel assembly at a time.

Monitoring Fuel and Waste Storage (Criterion 63)

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

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels.

Radiation monitors are provided to maintain surveillance over the waste release operation. The permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

There is a controlled ventilation system for fuel storage and waste treatment areas of the auxiliary building which discharges to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate a high-activity alarm on the control board annunciator.

Auxiliary shielding for the Waste Disposal System and its storage components was designed to limit the dose rate to levels not exceeding 0.75 mr/hr in normally occupied areas, to levels not exceeding 2.00 mr/hr in intermittently occupied areas and to levels not exceeding 15 mr/hr in limited occupancy areas.

The fuel handling mechanisms were designed so that it is unlikely that an accidental release of radioactivity can take place. These components are also contained within the fuel storage building which further reduces the chance of a "leak" and assists in maintaining the guidelines set up by 10 CFR 100. Furthermore, gamma radiation levels in the Fuel Storage

Building itself are continuously monitored by a local (R-5) Area Radiation Monitor. The monitoring serves to warn the operator of impending high radiation levels for such cases as low water level, contaminated water or improper handling of irradiated equipment or fuel elements. If the set point is reached, it is alarmed locally and in the Control Room.

Whenever the ventilation system is required to be in operation, the bypass damper around the charcoal filter must be manually closed. On a high radiation alarm, the following actions automatically take place:

1) Building ventilation supply fans are secured,
2) Dampers at ventilation supply fan close,
3) If open, rolling door closes,
4) Inflatable seals on main doors and truck doors are actuated (R-5 operability does not require this function, however), and
5) Exhaust fans continue to run.

Under these conditions, the maximum calculated in-leakage to the building (as a result of non-air tight construction) would be 20,000 cfm with a one-half inch of water negative pressure inside the building. Thus, there will be zero air leakage from the building proper, and the entire exhaust from the building will pass through roughing HEPA and charcoal filters before passing into the plant vent.

A spent fuel pit cooling loop which is a part of the Auxiliary Coolant System is provided to remove from the spent fuel pit the heat generated by the stored fuel elements. Both the water level and temperature are continuously monitored. High and low levels in the pit (6" above or below the 93' –8" normal) are alarmed in the Control Room, as is high temperature of the water in the pit (135 ^{0} F).

Two monitor tanks are provided to collect liquid wastes processed by the liquid waste disposal system that are suitable for direct release to the river. When a monitor tank is full, it will be isolated and the second tank will be placed in service. The isolated tank is then recirculated and a sample is taken.

The sample taken will be analyzed for gross activity. If the water is considered unsuitable for discharge, it will be returned to the waste holdup tank for reprocessing.

A direct measurement of the activity can be made by means of the radiation detector R-18 located in the monitor pumps' discharge line so that the liquid wastes can be monitored during both recirculation and discharge. The activity level will be indicated on the Waste Disposal Panel and in the Control Room (on the Radiation Monitoring System Cabinet). If the activity exceeds the high alarm setpoint, an alarm will be annunciated at the waste disposal panel, "WDS Liquid Monitor Hi Radiation." In addition, if the activity reaches the alarm point, the control valve RCV-018 in the waste release line will be tripped shut via an electrical interlock. This radiation detector thus

Ch.

2

CHAPTER 2

SITE AND ENVIRONMENT

2.1 SUMMARY OF CONCLUSIONS

This Chapter sets forth the site and environmental data which together formed the basis for many of the criteria for designing the facility and for evaluating the routine and accidental releases of radioactive liquids and gases to the environment. These data support the conclusion that there will be no undue risk to public health and safety due to plant operation. The strength of this conclusion rests with these data and the determinations (also included in this report) of several independent consultants, each speaking within a particular area of expertise - - health physics, demography, geology, seismology, hydrology or meteorology, as the case may be.

The task of evaluating the environmental characteristics of the area was facilitated by the fact that more than 12 years of studies and measurements of environmental characteristics were undertaken. For over twenty years, measurements have been made of the effects on the environment of releases from at least one operating nuclear power facility at the Indian Point Site.

Conservative projections have been made of the growth of population in the area and these projections have been taken into account in plant design and operation as to control the effects of accidents. Population estimates are presented in subchapter 2.4.

The census data for 1990 reveals that the population within a 10-mile radius of the site was approximately 238,043 whereas the 2000 estimated population is 564,200. The land is now zoned principally for residential and state park usage although there is some industrial activity and a little agricultural and grazing activity. The projections do not indicate that the land usage within this radius will shift appreciably during the period of plant operation.

Geologically, the site consists of a hard limestone formation in a jointed condition which provides a solid bed for the plant foundation. The bedrock is sufficiently sound to support any loads up to 50 tons per square foot, which is far in excess of any load imposed by the plant. Although it is hard, the jointed limestone formation is permeable to water. Thus, if water

from the plant should enter the ground (an improbable event since the plant is designed to preclude any leakage into the ground) it would percolate to the river rather than enter any ground water supply. Additional studies by the geology consultant, Thomas W. Fluhr, and examination of soil borings confirmed the above conclusions.

In the Hudson River, about 80,000,000 gallons of water flow past the plant each minute during the peak tidal flow. This flow provides additional mixing and dilution for liquid discharges from the facility. Plant design was based on the conservative assumption that the river water is used for drinking, thus radioactive discharges are reduced by dilution with ordinary plant effluent to concentrations that would be tolerable for drinking water. There is very little danger of flooding at the site.

Significant seismic activity in the Indian Point area is rare and no damage has resulted therefrom. As stated by the consultant on seismology, the site is "practically non-seismic" and is as safe as any area, at present known." Notwithstanding such assurance, the plant is designed to withstand an earth- quake of the highest intensity ever recorded in this area.

Meteorological conditions in the area of the site were determined during a two-year test program (1955 to 1957). The validity of these conclusions was verified by a test program completed in October 1970. The meteorological analysis also includes data from periods of November 26, 1969 through October 1, 1970, and January 1, 1970 through December 31, 1971. These data were used in evaluating the effects of gaseous discharges from the plant during normal operations and during a postulated Loss-of-Coolant Accident. In addition, data supplied by the U.S. Weather Bureau at the Bear Mountain Station, regarding the meteorological conditions during periods of precipitation, have been used to evaluate the rainout of fission gases into surface water reservoirs following a postulated Loss-of-Coolant Accident. The evaluations indicate that the site meteorology provides adequate diffusion and dilution of any released gases.

Environmental radioactivity has been measured at the site and surrounding area for nearly twenty years in association with the operation of Indian Point 1, and the construction and operation of Indian Point 2 and 3. These measurements are being continued and reported as dictated by the Technical Specifications and ODCM. The radiation measurements of fallout, water samples, vegetation, marine life, etc. have shown no perceptible post-operative increase in radioactivity due to plant operations. Noticeable increases in fallout have coincided with weapons testing programs and appear to be related almost entirely to those programs. The New York State Department of Health, in an independent two-year post-operative study[1], found that environmental radioactivity in the vicinity of the site is no higher than anywhere else in the State of New York.

Consultants who participated in the preparation of the various reports, mea-surements and conclusions appearing in this Chapter included Dr. Merril Eisenbud, then Director of Environmental Radiation Laboratory, Institute of

2.9 ENVIRONMENTAL MONITORING PROGRAM

2.9.1 General

A program to determine the environmental radioactivity in the vicinity of Indian Point Station was instituted in 1958, four years prior to the initial operation of Consolidated Edison's Indian Point Unit No. 1. The purpose of this survey was to determine the natural background radioactivity and to show the variations in the activities that may be expected from natural sources, fallout from bomb tests and other sources in the vicinity. This program has been continued to the present so that changes in the environment, resulting from station operations, could be accounted for. The results of these surveys are reported annually to the Nuclear Regulatory Commission.

In addition, the New York State Department of Health has conducted surveys throughout the State of New York since 1955, including extensive surveys in the vicinity of the Indian Point Station since 1958. In 1965 and 1966, they reported the findings in the vicinity of the Indian Point Station in two special reports. Since that time, their reporting has been on a statewide basis in quarterly bulletins and in annual reports.

In 1964, the New York University Medical Center began a research program on the ecology of the Hudson River. The New York University studies include the biology of the Hudson River, the distribution and abundance of fish in the river, pesticides and radio-ecological studies. The results of this program, supported by the United States Public Health Service, the New York State Department of Health, and the Consolidated Edison Company have been submitted in several program reports.

The various studies mentioned above included measurements of radioactivity in fresh water, river water, river bottom sediments, fish, aquatic vegetation, soil, vegetation and air in the vicinity of the Indian Point Station. The results of these monitoring programs have shown that the operation of the Indian Point Units 1, 2, and 3 have had no deleterious effects on the environment.

2.9.2 Survey Programs

The survey of environmental radioactivity in the vicinity of Indian Point Station provides an indication of the integrity of the in-plant radiation monitoring instrumentation and can reveal any buildup of long lived radionuclides.

By determining the activity of filterable air particulate, vegetation, drinking water and aboveground gamma fields, an indirect monitoring of discharges to the atmosphere is provided by the environmental survey program.

The effect of liquid effluents on the Hudson River is monitored by measuring the activity of the cooling water inlet to and discharge from the station, discharges from the plant, activity analysis of river shoreline soils and river fish and invertebrates.

A detailed description of the media sampled in accordance with plant Environmental Monitoring Program and the ODCM is given below:

Air Particulate and Organic Iodide

Concentration of radioactive particles in the air is measured weekly from 5 stations.

Membrane filters precede charcoal impregnated filters. The particulate filters are assayed for gross beta activity and are composited for quarterly gamma spectral analysis. Charcoal filters have gamma spectral analysis for I-131 performed weekly.

Reservoir Water

Drinking water is sampled monthly from an area reservoir. The water sampled is analyzed for gross beta activity, and for other nuclides via gamma spectral analysis. A quarterly composite sample is analyzed for tritium.

Hudson River Water

Continuous flow samples of the condenser inlet cooling water and discharge water are collected and composited. Samples are taken, at a frequency specified in the ODCM, from continuous samples and composited for a monthly gamma spectroscopy analysis, and for a quarterly tritium analysis.

Ch.

3

Specifically, a bank contains either one or two groups. Within a bank the group may be moved only sequentially one step at a time. Two banks may be moved simultaneously, e.g. banks C and D. However, group 1 in bank D may be moved together (one step), then group 2 in each bank simultaneously (one step). Therefore, no more than two groups can be moved together and this forms the basis of the assumption.

No single credible mechanical or electrical control system malfunction can cause a rod cluster to be withdrawn at a speed greater than 80 steps per minute (~50 inches per minute).

3.1.3 <u>Safety Limits</u>

The reactor is capable of meeting the performance objectives throughout core life under both steady state and transient conditions without violating the integrity of the fuel elements. Thus the release of unacceptable amounts of fission products to the coolant is prevented.

The limiting conditions for operation established in the Technical Specifications specify the functional capacity of performance levels permitted to assure safe operation of the facility.

Design parameters which are pertinent to safety limits are specified below for the nuclear, control, thermal and hydraulic, and mechanical aspects of the design.

<u>Nuclear Limits</u>

At full power (license application power) the nuclear heat flux hot channel factor F_Q^N, is not exceeded. For any condition of power level, coolant temperature and pressure which is permitted by the control and protection system during normal operation and anticipated transients the hot channel power distribution is such that the minimum DNB ratio is greater than the applicable limit and the linear fuel rating is less than 21 kW/ft. For any normal steady state operating condition, the maximum linear fuel rating does not exceed 6.24 x FQ kW ft, where FQ is the maximum value dictated by the Core Operating Limits Report (COLR).

Potential axial xenon oscillations are controlled with the control rods to preclude adverse core conditions. The protection system ensures that the nuclear core limits are not exceeded.

Fuel Enrichment Limits

Detailed nuclear analysis (refer to Reference 65 of Section 3.2) has been completed to demonstrate that the existing spent fuel storage racks can safely store fuel with initial enrichments up to 4.5 w/o U-235 provided they are done so as specified in Figures 9.5-2A, 9.5-2B, and 9.5-2C.

Control Bank Insertion Limits

The control bank insertion limits for D, C and B control banks were originally revised for Cycle 6 operation. The associated change in control bank insertion limits (refer to Reference 57 of Section 3.2) results in increased flexibility in core design, and a reduction in the calculated core peaking factor Fq at the bank insertion limit. The revised insertion limit curve for Cycle 10 is provided in Figure 3.2-39.

Reactivity Control Limits

The control system and the operational procedures provide adequate control of the core reactivity and power distribution. The following control limits are met:

1) A minimum hot shutdown margin as shown in the Technical Specifications is available assuming a 10% uncertainty in the control rod calculation (see Table 3.2-3).

2) This shutdown margin is maintained with the most reactive RCCA in the fully withdrawn position.

3) The shutdown margin is maintained at ambient temperature by the use of soluble poison.

Thermal and Hydraulic Limits

The reactor core was designed to meet the following limiting thermal and hydraulic criteria:

1) The minimum allowable DNBR during normal operation, including anticipated transients, is 1.30 for the W-3 correlation at system pressure >1000 psi, and 1.45 at low pressure (500 – 1000 psi). The minimum allowable DNBR for the WRB-1 correlation is documented in Table 14.1-0.

2) Fuel temperature will not exceed 4700° F during any anticipated operating condition.

The average temperature of the reactor coolant is increased with power level in the reactor. Since this change is actually a part of the power dependent reactivity change, along with the Doppler effect and void formation, the associated reactivity change must be controlled by rods. The largest amount of reactivity that must be controlled is at the end of life when the moderator temperature coefficient has its most negative value. The moderator temperature coefficient range is given in Table 3.2-1, line 42, while the cumulative reactivity change is shown in the first line of Table 3.2-2. By the end of the fuel cycle, the nonuniform axial depletion causes a severe power peak at low power. The reactivity associated with this peak is part of the power defect.

Operational Maneuvering Band

The control group is operated at full power within a prescribed band of travel in the core to compensate for periodic changes in boron concentration, temperature, or xenon. The band has been defined as the operational maneuvering band. When the rods reach either limit of the band, a change in boron concentration must be made to compensate for any additional change in reactivity, thus keeping the control group within the maneuvering band.

The greatest depth in the core to which the control banks may be inserted is known as the insertion limit. The maximum insertion limit on which the FSAR transient analyses are based is 23.5 percent D-Bank insertion, which corresponds to a D-Bank position of 176 steps at 100 percent power. The actual limit may be administratively decreased via a COLR change, to facilitate core design without having to reanalyze the FSAR transients.

The fully withdrawn bank position can vary within a few steps from the reference fully withdrawn condition from cycle to cycle. The Cycle 7 and 8 fully withdrawn position was 226 steps, and Cycle 9 through the current cycle is 230 steps.

Control Rod Bite

If sufficient boron is present in a chemically-shimmed core, the inherent operational control afforded by the negative moderator temperature coefficient is lessened to such a degree that the major control of transients resulting from load variations must be compensated for by control rods. The ability of the plant to accept major load variations is distinct from safety considerations, since the reactor would be tripped and the plant shut down safely if the rods could not follow the imposed load variations. In order to meet required reactivity ramp rates resulting from load changes, the control rods must be inserted a given distance into the core. The reactivity worth of this insertion has been defined as control rod bite.

The reactivity insertion rate must be sufficient to compensate for reactivity variation due to changes in power and temperature caused either by a ramp load change of five percent per minute, or by a step load change of ten percent. An insertion rate of $4 \times 10^{-5}\ \Delta p$ per second is determined by the transient analysis of the core and plant to be adequate for the most adverse combinations of power and moderator coefficients. To obtain this minimum ramp rate, one control bank of rods had to remain inserted at least 13 percent into the core at the Cycle 1 beginning-of-life. The reactivity associated with this bite was 0.03 percent.

Xenon Stability Control

The control rods are capable of suppressing xenon induced power oscillations in the axial direction, should they occur. Out-of-core instrumentation was provided to obtain necessary information concerning power distribution. This instrumentation is adequate to enable the operator to monitor and control xenon induced power oscillations. Extensive analyses, with confirmation of methods by spatial transient experiments at Haddam Neck, have shown that any induced radial or diametral xenon transients would die away naturally.[2] A full discussion of axial xenon stability control can be found in Reference 3.

Excess Reactivity Insertion Upon Reactor Trip

The control requirements are nominally based on providing 1.3 percent shutdown at hot, zero power conditions with the highest worth rod stuck in its fully withdrawn position, or to prevent return to criticality following a credible steam line break, whichever is the more limiting. The condition where excess reactivity insertion is most critical is at the end of a cycle when the steam line break accident is considered. The excess control available at the Cycle 1 end-of-cycle, hot zero power condition with the highest worth rod stuck out, allowing a 10% margin for uncertainty in control rod worth is shown in Table 3.2-3.

Calculated Rod Worths

The complement of 53 full length control rods, arranged in the pattern shown in Figure 3.2-1 meets the shutdown requirements. Table 3.2-3 lists the calculated worths of this rod configuration for beginning-of-life, and end-of life, for Cycles 1 and the current cycle.

The first category comprises uncontrolled rod withdrawal (with rods moving in the normal bank sequence). Also included are motions of the banks below their insertion limits, which could be caused, for example, by uncontrolled dilution or primary coolant cooldown. Power distributions are calculated throughout these occurrences assuming short-term corrective action, that is, no transient xenon effects are considered to result from the malfunction. The event is assumed to occur from typical normal operating situations which include normal xenon transients. It is further assumed in determining the power distributions, that total core power level will be limited by reactor trip to below 118 percent. Since the study is to determine protection limits with respect to power and axial offset, no credit is taken for trip setpoint reduction due to flux difference. The peak power density which can occur in such events, assuming reactor trip at or below 118 percent, is less than that required for centerline melt, including uncertainties and densification effects.

The second category assumes that the operator mispositions the rod bank in violation of the insertion limits and creates short term conditions not included in normal operating conditions.

The third category assumes that the operator fails to take action to correct a flux difference violation. The resulting F_Q is multiplied by 102 percent power including an allowance for calorimetric error. It should be noted that a reactor overpower accident is not assumed to occur coincident with an independent operator error.

Analyses of possible operating power shapes show that the appropriate hot channel factors F_Q and $F^N_{\Delta H}$ for peak local power density and for DNB analysis at full power are the values addressed in the COLR.

F_Q can be increased with decreasing power as shown in the Technical Specifications. Increasing $F^N_{\Delta H}$ with decreasing power is permitted by the DNB protection setpoints and allows radial power shape changes with rod insertion to the insertion limits. It has been determined that, provided the above conditions 1) through 4) are observed, the COLR limits are met.

When a situation is possible in normal operation which could result in local power densities in excess of those assumed as the precondition for a subsequent hypothetical accident but which would not itself cause fuel failure, administrative controls and alarms are provided for returning the core to a safe condition.

Reactivity Coefficients

The response of the reactor core to plant conditions or operator adjustments during normal operation, as well as the response during abnormal or accidental transients, is evaluated by means of a detailed plant simulation. In these calculations, reactivity coefficients are required to couple the response of the core neutron multiplication to the variables which are set by conditions external to the core. Since the reactivity coefficients change during the life of the core, a range of coefficients is established to determine the response of the plant throughout life and to establish the design of the Reactor Control and Protection Systems.

Moderator Temperature Coefficient

The moderator temperature coefficient in a core controlled by chemical shim is less negative than the coefficient in an equivalent rodded core. One reason is that control rods contribute a negative increment to the coefficient, and in a chemical shim core the rods are only partially inserted. Also, the chemical poison density is decreased with the water density upon an increase in temperature. This gives rise to a positive component of the moderator temperature coefficient due to boron being removed from the core. This is directly proportional to the amount of reactivity controlled by the dissolved poison.

In order to reduce the dissolved poison requirement for control of excess reactivity, burnable poison rods have been incorporated in the core design. The result is that changes in the coolant density will have less effect on the density of poison, and the moderator temperature coefficient will be reduced.

The burnable poison for Cycle 1 was in the form of borated Pyrex glass rods clad in stainless steel. There were 1434 of these borated Pyrex glass rods in the form of clusters distributed throughout the initial core in vacant rod cluster control guide tubes, as illustrated in Figures, 3.2-6 through 3.2-7F. Information regarding research, development and nuclear evaluation of the burnable poison rods can be found in Reference 1. These rods initially controlled 10% $\Delta\rho$ of the installed excess reactivity and their insertion into the core resulted in a reduction of the initial hot zero power boron concentration in the coolant to 1330 ppm. The moderator temperature coefficient is negative at operating conditions with burnable poison rods installed. Subsequent cycles utilized B_4C in $AL_2 0_3$ in wet annular burnable absorbers (WABA) and ZrB_2 in Integral Fuel Burnable Absorbers (IFBA) and also the already mentioned Pyrex rods.

Ch.

4

For the combination of normal and Design Basis Earthquake loadings the stresses in the support structures were limited to values as necessary to assure their integrity and to maintain the stresses in the Reactor Coolant System components within the allowable limits as previously established.

4.1.5 Cyclic Loads

All components in the Reactor Coolant System were designed to withstand the effects of cyclic loads due to reactor system temperature and pressure changes. These cyclic loads are introduced by normal unit load transients, reactor trip, and startup and shutdown operation. The number of thermal and loading cycles used for design purposes and the bases thereof are given in Table 4.1-8. During unit startup and shutdown, the rates of temperature and pressure changes are limited as indicated in Section 4.4.1. The cycles are estimated for equipment design purposes (40 year life) and are not intended to be an accurate representation of actual transients or actual operating experience.

To provide the necessary high degree of integrity for the equipment in the Reactor Coolant System, the transient conditions selected for equipment fatigue evaluation were based on a conservative estimate of the magnitude and frequency of the temperature and pressure transients resulting from normal operation, normal and abnormal load transients. To a large extent, the specific transient operating conditions considered for equipment fatigue analyses were based upon engineering judgement and experience. Those transients were chosen which are representative of transients to be expected during plant operation and which are sufficiently severe or frequent to be of possible significance to component cyclic behavior.

Clearly it is difficult to discuss in absolute terms the transients that the plant will actually experience during the 40 years operating life. For clarity, however, each transient condition is discussed in order to make clear the nature and basis for the various transients.

1. Heatup and Cooldown

   For design evaluation, the heatup and cooldown cases were represented by continuous heatup or cooldown at a rate of 100 F per hour which corresponds to a heatup or cooldown rate under abnormal or emergency conditions. The heatup occurs from ambient to the no load temperature and pressure condition and the cooldown represents the reverse situation. In actual practice, the rate of temperature change of 100 F per hour will not be usually attained because of the Over Pressure Protection System (OPS) discussed in Section 4.3 and other limitations such as:

a) Criteria for protection against non-ductible failure which establish maximum permissible temperature rates of change, as a function of plant pressure and temperature

b) Slower initial heatup rates when using pumping energy only

c) Interruptions in the heatup and cooldown cycles due to such factors as drawing a pressurizer steam bubble, rod withdrawal, sampling, water chemistry and gas adjustments.

The heatup and cooldown rates, administratively imposed by plant operating procedures, are limited to 50 F per hour for normal operation. Ideally, heatup and cooldown would occur only before and after refueling. In practice, additional scheduled and unscheduled plant cooldowns may be necessary for plant maintenance.

2. <u>Unit Loading and Unloading</u>

The unit loading and unloading cases were conservatively represented by a continuous and uniform ramp power change of 5% per minute between no load and full load. The reactor coolant temperature varies with load as prescribed by the temperature control system. The number of each operation was specified at 14,500 times or once per day for the 40-year plant design life. In practice, the plant is operated at base load.

3. <u>Step Increase and Decrease of 10%</u>

The ±10% step change in load demand is a control transient which is assumed to be a change in turbine control valve opening which might be occasioned by disturbances in the electrical network into which the plant output is tied. The Reactor Control System was designed to restore plant equilibrium without reactor trip following a ±10% step change in demand. The turbine load power range for automatic reactor control initiated from nuclear plant equilibrium conditions, is in the range between 15% and 100% of full load. In effect, during load change conditions, the Reactor Control System attempts to match turbine and reactor outputs so that peak reactor coolant temperature is minimized. Concurrently, the reactor coolant temperature is restored to its programmed set point at a sufficiently slow rate to prevent excessive pressurizer pressure decrease.

Following a step load decrease in turbine load, the secondary side steam pressure and temperature initially increase since the decrease in nuclear power lags behind the step decrease in turbine load. During the same increment of time, the Reactor Coolant System average temperature and pressurizer pressure also initially increase.

Because of the power mismatch between the turbine and reactor, the increase in reactor coolant temperature is ultimately reduced from its peak value to a value below its initial equilibrium value at the inception of the transient. The reactor coolant average temperature set point change is made as a function of turbine-generator load as determined by first stage turbine pressure measurement. The pressurizer pressure will also decrease from its peak pressure value and follow the rector coolant's decreasing temperature trend. At some point during the decreasing pressure transient, the saturated water in the pressurizer begins to flash, which reduces the rate of pressure decrease.

Subsequently, the pressurizer heaters come on to restore the plant pressure to its normal value.

Following a step load increase in turbine load, the reverse situation occurs, i.e., the secondary side steam pressure and temperature initially decrease and the reactor coolant average temperature and pressure initially decrease. The control system automatically withdraws the control rods to increase core power. The decreasing pressure transient is reversed by actuation of the pressurizer heaters and eventually the system pressure is restored to its normal value. The reactor coolant average temperature is raised to a value above its initial equilibrium value at the beginning if the transient.

The number of each operation was specified at 2000 times or 50 per year for the 40-year plant design life.

4. <u>Large Step Decrease in Load</u>

This transient applies to a step decrease in turbine load from full power of such magnitude that the resultant rapid increase in reactor coolant average temperature and secondary side steam pressure and temperature automatically initiates a secondary side steam dump that prevents a reactor shutdown or the lifting of steam generator safety valves.

The number of occurrences of this transient was specified at 200 times or 5 per year for the 40-year plant design life.

5. <u>Reactor Trip from Full Power</u>

A reactor trip from full power may occur for a variety of causes resulting in temperature and pressure transients in the Reactor Coolant System and in the secondary side of the steam generator. This is the result of continued heat transfer from the reactor coolant in the steam generator. The transient continues until the rector coolant and steam generator secondary side temperatures are in equilibrium at zero power conditions. A continued

supply of feedwater and controlled dumping of secondary steam removes the core residual heat and prevents the stem generator safety valves from lifting. The reactor coolant temperature and pressure undergo a rapid decrease from full power values as the reactor protection system causes the control rods to move into the core.

The number of occurrences of this transient was specified at 400 times or 10 per year for the 40-year plant design life.

6. Hydrostatic Test Conditions

The pressure tests outlined below apply to field pressure tests conducted on the erected Reactor Coolant System. The number of tests given below does not include any allowance for pressure tests conducted on a specific component in the manufacturer's shop in accordance with vessel code requirements.

a. Primary Side Hydrostatic Test before Initial Startup at 3110 psig

This hydrostatic test was performed at a minimum water temperature of 100 F, imposed by a reactor vessel material NDTT value of 100 F at beginning of life, and a maximum test pressure of 3110 psig. In this test, the primary side of the steam generator was pressurzied to 3110 psig coincident with the secondary side pressure of 0 psig. The Reactor Coolant System was designed on the basis of 5 cycles of this hydro test.

b. Recator Coolant System Leakage Test

This test is performed at normal operating pressure following each refueling outage prior to startup in accordance with ASME Section XI. Additional tests are performed following repairs, replacements or modifications of the RCS in accordance with ASME Section XI.

For conservatism, however, 125% of operating speed was selected as the design speed for the primary coolant pumps. For the overspeed condition, which would not persist for more than 30 seconds, pump operating temperatures would remain at about the design value.

d) Bursting Speed

Bursting speed of the flywheels has been calculated on the basis of Robinson's results[1] to be 3900 rpm, more than three times the operating speed. This is confirmed using Griffith-Irwin theory.[2]

Steam Generators

The pressure boundary integrity of the steam generator, including the primary to secondary pressure boundary is assured by compliance of the design, fabrication, analysis, inspection, and testing activities with the criteria and requirements of the ASME Boiler and Pressure Vessel Code. The stress report for the Model 44F steam generators currently installed in Indian Point Unit 3 included an evaluation for faulted conditions including large break LOCA and steam line break (loss of secondary pressure). The stress intensities calculated for these conditions are less than the applicable limits from the ASME Code. The criteria and requirements of Section III of the ASME Code (1965 edition, Winter 1965 Addenda) were used for the evaluation.

The evaluation of the stress intensity levels in the tubesheet and channel head for these faulted conditions is based on an evaluation of interactions of the complex structure of the channel head, tubesheet and lower shell. A finite element computer program is used for the evaluation. The structure is modeled in terms of discrete elements with loading and boundary conditions applied to these elements. The system of simultaneous linear equations resulting from the modeling is solved to determine the stress conditions. This method of stress analysis is well established for reactor coolant system components. For the tubesheet calculations, the guidelines of Article I-9, ASME Code, Section III were used to calculate the ligament efficiency based on nominal pitch dimension and maximum hold dimensions.

The postulated rupture of a primary pipe is assumed to impose a maximum pressure differential of 1100 psi across the tubes and tubesheet. The maximum local primary membrane plus primary bending stress in the tubesheet under these conditions is 21,620 psi. This is well below the ASME Code stress intensity allowable of 84,000 for this condition. The stress intensities in the channel head, channel head to tubesheet weld, and the tubesheet to lower shell weld have lower values and are well below applicable limits for this condition.

The postulated rupture of a secondary pipe is assumed to impose a maximum pressure differential of 2485 psi across the tubes and tubesheet. The maximum local primary membrane plus primary bending stress in the tubesheet under these conditions is 54,650 psi. This is well below the ASME Code stress intensity allowable of 84,000 for this condition (Tables 4.3-3 and 4.3-4). The stress intensities in the channel head, channel head to tubesheet weld, and the tubesheet to lower shell weld have lower values and are well below applicable limits for this condition.

The stress intensity in the tubes has also been considered for the postulated faulted conditions. In addition to the pressure differential stresses, LOCA blowdown forces result in a bending load on the tubes. The fatigue due to vibration of the tubes during a steam line break does not need to be considered in the evaluation. The requirements of the ASME Code are met for these faulted conditions.

In addition to analytical results, results of destructive pressure testing or representative tubes demonstrates a factor of safety against tube collapse due to external pressure. The results of the pressure testing have been used to calculate a collapse pressure for tubes of the size and material in the Model 44F steam generators. The lower bound collapse pressure for the tubes in the Indian Point steam generators is 2369 psi considering tube ovality, tube wear, and tube corrosion.

The structural evaluation of the tubes uses an allowance of 2 mils of uniform wall thinning. This value is based on published values and operating experience for corrosion and erosion-corrosion. The plugging limit for indication of tube degradation is based on the requirements of IWB-3521 of the ASME Boiler and Pressure Vessel Code Section XI or an analysis meeting the requirements of Regulatory Guide 1.121 and not on the allowance for uniform thinning in the structural evaluation.

The loading conditions considered include the maximum potential earthquake loading conditions superimposed on the loss of secondary pressure effects. The dynamic effects of the fluid and the acceleration of the steam generator result in a small increase in the equivalent pressure loading compared to the base pressure differential. The stress intensity for the combined loading condition is well below limits.

The fluid dynamic load on the tube support plate for the steam line break conditions has been considered. The analysis has determined that the tube supports will be restrained without deformation of the tubes.

In addition, the secondary side of the steam generator shall not be pressurized above 200 psig if the temperature of the steam generator is below 70 °F.

4.3.2 Reliance on Interconnected Systems

The principal heat removal systems which are interconnected with the Reactor Coolant System are the Steam and Power Conversion, the Safety Injection and the Residual Heat Removal Systems. The Reactor Coolant System is dependent upon the steam generators and upon the steam, feedwater, and condensate system for decay heat removal under normal operating conditions until RHR can be placed in service (reactor coolant temperature between 250° F and 350° F). The layout of the system ensures the natural circulation capability to permit adequate core cooling following a loss of all main reactor coolant pumps.

The flow diagram of the Steam and Power Conversion System is shown on Figure 10.2-1. In the event that the condensers are not available to receive the steam generated by residual heat, the water stored in the feedwater system may be pumped into the steam generators and the resultant steam vented to the atmosphere. The Axiliary Feedwater System will supply water to the steam generators in the event that the main feedwater pumps are inoperative.

The Safety Injection System is described in Section 6.2. The Residual Heat Removal System is described in Section 9.3.

4.3.3 System Integrity

A complete stress analysis, which reflected consideration of fall design loadings detailed in the design specification, was prepared by the manufacturer. The analysis showed that the reactor vessel, steam generators, reactor coolant pump casings and pressurizer complied with the stress limits of Section III of the ASME Code. A similar analysis of the piping showed that it complied with the stress limits of the applicable USAS Code.

As part of the design control on materials, Charpy V-notch toughness tests were run on all ferritic material used in fabricating pressure parts of the reactor vessel, steam generators and pressurizer to provide assurance that hydrotesting and operation will be in the ductile region at all times. In addition, dropweight tests were performed on the reactor vessel plate material.

As an assurance of system integrity, all components in the system were hydrotested at 3110 psig prior to initial operation. Replacement steam generators were shop tested on the primary side at 3107 psig, and hydrotested after installation in accordance with ASME Section XI requirements.

A summary of Charpy V-notch and dropweight test results for the reactor vessel plates and forgings is given in Section 4.4.

Furnace Sensitized Components

The following pressure or strength-bearing stainless steel component parts in the Reactor Coolant System have become furnace-sensitized during the fabrication sequence:

Reactor Vessel

Eight primary nozzle safe ends (forgings) which were overlaid in the field with stainless steel weld metal

Steam Generators

Two primary nozzle safe ends per generator – weld metal buttering

Pressurizer

All nozzle safe ends (forgings) in top and bottom heads.

Vibration and Cycle Loads

Vibration loads were considered in the design for the reactor internals, the steam generator tube bundles, and the reactor coolant pipe. Reactor coolant pump vibration is insignificant. Instrumentation is provided to check the vibration level of these pumps if an abnormal condition is suspected.

Reactor Internals

Model tests of the Indian Point 3 reactor internals were performed for normal operating and transient conditions. Results of the combined analytic and experimental work were factored into the design.

Predicted stresses and deflections were in agreement with tests on reactors having similar internals design. The results of the vibration tests performed on the Ginna rector (reported in WCAP-7408-L, Westinghouse proprietary report) [18] confirmed that the tests agree very closely with the predicted performance and margins. A more extensive testing program was performed during pre-operational testing for Indian Point 2.

Allowable stress amplitude for flow induced vibration was established on the basis of the material fatigue properties (endurance limit of 20,000 psi for 10^{10} cycles). Since infinite cycle fatigue was a criterion, no limits were then necessary for frequency. Displacement amplitudes for reactor internals vibration were not governing; stress limits were more restrictive.

An analysis of the dynamic response of the Indian Point 2 internals under seismic and blowdown loads was made. Allowable criteria were established and stresses and deflections were determined to assure that seismic and blowdown loads will not prevent core shutdown or will not interfere with the effectiveness of the emergency core cooling system (reported in detail in WCAP-7822, Westinghouse non-proprietary report).[15] This analysis applies directly to Indian Point 3.

Steam Generators

a) Tube Vibration Analysis

In the design of Westinghouse Model 44F steam generators used in Indian Point 3, the possibility of tube degradation due to either mechanical or flow-induced excitation was considered. This evaluation included detailed analysis of the tube support systems as well as an extensive research program with tube vibration model tests.

4.4.3 Maximum Pressure

The Reactor Coolant System serves as a barrier preventing radionuclides contained in the reactor coolant from reaching the atmosphere. In the event of a fuel cladding failure the Reactor Coolant System is the primary barrier against the uncontrolled release of fission products. By establishing a system pressure limit, the continued integrity of the Reactor Coolant System is assured. Thus, the safety limit of 2735 psig (110% of design pressure) has been established. This represents the maximum transient pressure allowable in the Reactor Coolant System under the ASME Code, Section III. Reactor Coolant System pressure settings are given in Table 4.1-1.

4.4.4 System Minimum Operating Conditions

Minimum operating conditions for the Reactor Coolant System for all phases of operation are given in the Technical Specifications and Technical Requirements Manual.

References

1) Hazeton, W. S., S. L. Anderson and S. E. Yanichko, "Basis for Heatup and Cooldown Limit Curves," WCAP-7924, July 1972.

2) Yanichko, S. E. and J. A. Davidson, "Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-8475, January 1975.

3) Davidson, J. A., S. L. Anderson and W. T. Kaiser, "Analysis of Capsule T from the Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-9491, April 1979.

4) Yanichko, S. E. and Anderson, "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit 3 Reactor Vessel Radiation Surveillance Program, Volume 1," WCAP-10300-1, March 1983.

5) Kaiser, W. T., "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 2," WCAP-10300-2, March 1983: Heatup and Cooldown Curves for Normal Operation.

6) Yanichko, S. E., "Analysis of Capsule Y From the Power Authority of the State of New York Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program, Volume 3," WCAP-10300-3, March 1983: J-Integral Testing of IX-WOL Fracture Mechanics Specimens.

7) Yanichko, S. E., "Analysis of Capsule Z From the New York Power Authority Indian Point Unit No. 3 Reactor Vessel Radiation Surveillance Program," WCAP-11815, March 1988.

8) Regulatory Guide 1.99, Rev. 2

9) 10 CFR 50.61 "Fracture Toughness Requirements for Protection Against Pressurized Thermal Shock Events," Federal Register, Vol. 56, No. 94, Effective May 15, 1991.

10) "Final Report on Pressure Temperature Limits for Indian Point 3 Nuclear Power Plant," ABB Combustion Engineering, July 1990.

11) Memorandum REC:92-003, dated February 10, 1992, F. Gumble to P. Kokolakis

12) "Indian Point Unit 3 Section XI Enable Temperatures for 13 and 15 EFPY," ABB Combustion Engineering, August 14, 1997.

Non-destructive testing was performed by one of several methods, as specified in Section XI and its applicable reference:

1) Visual Examination

   a) Direct Visual
   b) Remote Visual
   c) Indirect Visual

2) Surface Examination

   a) Magnetic Particle
   b) Liquid Penetrant

3) Volumetric Examination

   a) Radiographic
   b) Ultrasonic

Test personnel were qualified in accordance with code requirements.

Pre-Service Inspection

Section XI, IS-232 required pre-operational examination of essentially 100% of the pressure containing welds within the reactor coolant system boundary.

The plant components were examined in accordance with the requirements wherever it was possible and practical to do so in order to provide base line data for subsequent in-service inspections.

The pre-service examination for the original plant components was performed at the plant site after the components had been installed. With the exception of the reactor coolant pipe to channel head weld, which received a pre-service examination after the replacement steam generators were installed, pre-service examination of replacement steam generator pressure boundary welds was performed at the manufacturer's shop. Primary and secondary side hydrostatic tests were performed after installation. Personnel qualification, equipment and records met the requirements of applicable codes. Onsite examinations were necessarily limited by the design and accessibility restrictions of the plant.

In-Service Inspection

Operational examinations as set forth in ASME Section XI are performed to the fullest extent practical at the required intervals. When Reactor Coolant System modifications or repairs have been made which involve new strength welds on components, the new welds will meet the requirements of ASME Section XI. Also, the reactor coolant system shall be tested for leakage prior to plant startup following each refueling outage.

The structural integrity of the Reactor Coolant System is maintained at the level required by the original acceptance standards throughout the life of the plant. Any evidence resulting from the inspections required by the ISI Program and indicating that potential defect implications have initiated or enlarged are investigated, including evaluation of comparable areas of the Reactor Coolant System.

Non-destructive test methods, personnel, equipment and records conform to the requirements of ASME Section XI.

The use of conventional non-destructive, direct visual and remote visual test techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component. Because of these limitations on access to the reactor vessel, several features were incorporated into the design and manufacturing procedures in preparation for non-destructive test techniques as they become available:

1) Shop ultrasonic examinations were performed on all thermally clad surfaces to an acceptance and repair standard which assures an adequate cladding bond to allow later ultrasonic testing of the base metal. Size of cladding bonding defect allowed was ¼" x ¾"

2) The design of the reactor vessel shell in the core area is a clean, uncluttered cylindrical surface to permit future positioning of test equipment without obstruction

3) To establish baselines for Post-Operational Ultrasonic Testing of the Reactor Vessel, during the manufacturing stage selected areas of the reactor vessel were ultrasonic tested and mapped to facilitate the in-service inspection program.

   The areas selected for ultrasonic testing mapping included:

   a) Vessel flange radius, including the vessel flange to upper shell weld

   b) Middle shell course

   c) Lower shell course above the radial core supports

   d) Exterior surface of the closure head form the flange knuckle to the cooling shroud

   e) Nozzle to upper shell weld

   f) Middle shell to lower shell weld

   g) Upper shell to middle shell weld.

The pre-operational ultrasonic testing of these areas was performed after hydrostatic testing of the reactor vessel.

A qualified inspector employed by an insurance company authorized to write boiler and pressure vessel insurance certified all examinations.

Means of access to the Reactor Coolant Pressure Boundary were provided as necessary for the surveillance programs as detailed in the ISI Program. This inspection program is in compliance with Section XI of the ASME Code for in-service inspection of nuclear reactor coolant systems.

During the design phase, careful consideration was given to provide access for both visual and non-destructive in-service inspections of the reactor coolant primary and associated auxiliary systems and components within the boundaries established in accordance with the Section XI Code.

Specific provisions made for inspection access in the design of the reactor vessel, system layout and other major primary coolant components were:

1) All reactor internals are completely removable. The tools and storage space required to permit reactor internals removal for these inspections are provided

2) The reactor vessel shell in the core areas was designed with a clean, uncluttered cylindrical inside surface to permit future positioning of test equipment without obstruction

3) The reactor vessel cladding was improved in finish by grinding to the extent necessary to permit meaningful examination of the vessel welds and adjacent base metal in accordance with the Code

4) The cladding to base metal interface was ultrasonically examined to assure satisfactory bonding to allow the volumetric inspection of the vessel welds and base metal from the vessel inside surface

5) The reactor closure head is stored in a dry condition on the operating deck during refueling, allowing direct access for inspection

6) All reactor vessel studs, nuts, and washers are removed to dry storage during refueling, allowing inspection in parallel with refueling operations

7) Access holes were provided in the core barrel flange, allowing access for the remote visual examination of the clad surface of the vessel without removal of the lower internals assembly

8) Removable plugs were provided in the primary shield, providing limited access for inspection of the primary nozzle safe-end welds

9) Manways were provided in the steam generator channel head to provide access for internal inspection

10) A Manway was provided in the pressurizer top head to allow access for internal inspection

11) The insulation covering all component and piping welds 6 inches in diameter and larger and covering the adjacent base metal was designed for ease of removal and replacement in areas where external inspection is planned

12) Removable plugs were provided in the primary shield concrete above the main coolant pumps to permit removal of the pump motor and to provide internal inspection access to the pumps.

The Indian Point 3 reactor vessel was built to the 1965 edition of the ASME Code Section III and all addenda through the Summer 1965 issue. ASME Section XI in-service inspection was not a requirement at that time. However, accessibility and techniques are available for inspecting all welds requiring inspection by ASME Section XI on the vessel, except for the closure head dome and bottom head dome circumferential welds and the control rod mechanism housing and bottom instrumentation tube attachment welds which were completed prior to the issuance of Section XI. Particular design improvements applied to the reactor vessel to facilitate in-service inspection include Items 1) through 4) and 7) above.

The data and results of the pre-operational examination serve as baseline data for the in-service inspection program.

In-service inspection of seismic Class I pressure retaining components, such as vessels, piping, etc. within the Reactor Coolant Pressure Boundary is performed in accordance with Section XI of the ASME Code, 1983, Edition and Addenda up through the Summer 1983 issue with certain exceptions whenever specific relief is granted by the NRC. Details of the In-service Inspection Program are contained in the Weld and Support In-service Inspection Plan.

The engineered safety features, the reactor shutdown systems, the cooling water systems, and the radioactive waste treatment systems which are necessary for plant operation are provided as redundant systems. This redundancy provides the capability for system and/or component outage (per Technical Specification requirements) to perform operability tests/checks or repair/maintenance. Periodic testing is in accordance with Technical Specification requirements and the IP3 Pump and Valve Testing Plan. The Pump and Valve Testing Plan is in accordance with ASME Section XI with certain exceptions whenever specific relief is granted by the NRC.

B. Inspection

Wrought material was examined by UT and PT using Section III acceptance standards.

Reactor Coolant System Construction

All primary piping and fittings were given a solution annealing treatment consisting of heating to 1900 – 1950 F, holding 1 hour per inch of thickness and water quenching. This assured that the material would not be sensitized.

Main coolant pipe welds are of type 308L or 316 stainless steel. Welding was performed during original plant construction by the manual metal arc process after the root pass was completed using an insert followed by three layers using the manual gas shielded tungsten arc process. The maximum energy input possible with the manual metal arc process is on the order of 20,000 joules per linear inch of weld. With the large heat sink available in this thick walled pipe (2.375 to 3.00"), and the interpass temperature control of 350 F maximum, there was no sensitization of the solution treated pipe during welding.

Comparable welding controls to avoid primary piping sensitization were also employed during steam generator replacement, however, automatic gas metal arc welding processes were used after the root and hot passes were manually completed. The use of inserts was not required during the reconnection of primary piping to the replacement steam generators.

Venting provisions were made at high points throughout the Reactor Coolant System to relieve entrapped air when the system is filled and pressurized. Principally, vents were installed on the reactor vessel head, the pressurizer, and the reactor coolant pumps. Additional vents are available on the control rod drive mechanisms, on instruments, and on a number of connecting pipes. For normal venting of the Reactor Coolant System, only the principal venting points are utilized. The amount of oxygen which could be trapped in the remaining small volumes becomes negligible as the system is pressurized and the oxygen is scavenged by the hydrazine, specifically added for this purpose prior to operation. During operation, the oxygen levels are kept low consistent with water chemistry requirements as described in the Technical Requirements Manual.

Reactor Coolant System Operational Stresses

To avoid unusual stresses in areas where nozzle safe ends are joined to the piping, precautions were taken to eliminate unnecessary stresses due to erection of the various components of the Reactor Coolant System. The primary coolant system piping closure pieces are two pipe fitting subassemblies located between the steam generator and the primary coolant pump. The 40 degree elbow of the loop piping was first installed on the steam generator outlet nozzles. Then the gap to be closed by the closure pieces was physically measured between the 40 degree elbow outlet and the inlet nozzle of the pump. These measured dimensions for each individual loop were compensated and adjusted for the expected field weld shrinkage. The resulting net true dimensions were then transmitted to the pipe shop fabricator who prepared the final closure pipe subassemblies for each primary coolant loop. Upon welding these specially dimensioned pipe subassemblies in place, the primary coolant system closure was accomplished for each loop in a condition which was free from cold spring. During steam generator replacement, customized steam generator primary nozzle coordinates, temporary pipe restraints, mechanical and optical templating methods, and precision machining were all employed to ensure restoration of the RCS to its original configuration.

As a precaution that the behavior of the Reactor Coolant System during operating conditions was as predicted, measurements were made during incremental temperature increases during the hot functional test. The measurements were made to check the movement of the components at temperature and pressure to insure interferences were not present. The data taken during the test were compared with the flexibility analysis predications and evaluated.

Inservice Inspection Capability

As a final check on the adequacy of the precautions taken to avoid any Reactor Coolant System failure as a result of severely sensitized stainless steel, a post-operational inspection plan was developed for the nozzle safe ends within the Reactor Coolant System Boundary. The pressurizer and steam generator stainless steel safe ends which were subjected to the furnace atmosphere during final stress relief are accessible for visual, surface and volumetric inspection upon removal of the insulation at each safe end. The reactor vessel safe ends which were subjected to the furnace atmosphere are accessible for limited inspection by removal of the special access plugs provided in the primary concrete just above each nozzle. Upon removal of these plugs and the insulation of the safe end, approximately 120 degrees of the top segment of the safe ends are accessible for direct visual and surface examination.

A specially designed in-vessel, remote, ultrasonic, inservice inspection tool was developed which can be affixed to the upper vessel flange after removal of the head. This tool is intended for ultrasonic examination of the vessel circumferential and longitudinal welds, nozzle-to-vessel welds and nozzle-to-safe-end welds. Some of these examinations utilizing this in-vessel tool were performed in the 1979 refueling outage. No indications were revealed by this testing.

Ch. 5

5.1.7.2 Pre-Operational Tests

All penetrations, and the welds joining these penetrations to the containment liner and the liner seam welds, were designed to provide a double barrier which can be continuously pressurized at a pressure higher than the calculated peak accident response pressure of the containment. This blocks potential sources of leakage with a pressurized zone and at the same time provides a means of monitoring the leakage status of the containment which is more sensitive to changes in the leakage characteristics of these potential leakage sources. Certain liner welds are no longer continuously pressurized. Therefore, the leakage status of these welds is no longer continuously monitored. The integrity of these welds is verified by integrated leak rate testing.

After the Containment Building was complete with liner, concrete structures, and all electrical and piping penetrations, Equipment Hatch and Personnel Lock in place, the following tests were performed:

1) Strength Test:

   A pressure test was made on the completed building using air at 54 psig. This pressure was maintained on the building for a period of at least one hour. During this test, measurements and observations were made to verify the adequacy of the structural design. For a description of observations, cracks, strain gauges, etc., refer to the Containment Report, Appendix 5A.

2) Integrated Leakage Rate Tests:

   Integrated leakage rate tests were performed on the completed building using the absolute method. These leakage tests were performed with the double penetration and weld channel zones open to the containment atmosphere.

3) Sensitive Leak Rate Test:

   After it had been assured that there were no defects remaining from construction, a sensitive leak rate test was conducted. The sensitive leak rate test included only the volume of the weld channels and double penetrations. This test is considered more sensitive than the integrated leakage rate test, as the instrumentation used permits a direct measurement of leakage from the pressurized zones. The sensitive leak rate test was conducted with the penetrations and weld channels at a minimum pressure greater than the calculated peak accident pressure and with the Containment Building at atmospheric pressure. The leak rate for the double penetrations and weld channel zones was equal to or less than 0.2% of the containment free volume per day.

In order to verify that the structural response of the Containment to pressure loads is in accordance with design assumptions and to provide assurance that the structure was constructed in accordance with the design to resist pressure loads, a Structural Integrity Test (SIT) was performed.

Readings and measurements were taken at 0 psig, 12 psig, 21 psig, 41 psig and 54 psig (the latter is 115% of the design pressure of 47 psig) during pressurization, and at 41, 18, 21, 41, and 0 psig during depressurization.

The following gross deformation measurements were taken during the SIT using invar wire extensometers. This provided a means for taking all measurements inside the containment structure thus eliminating effects of weather and temperature. All results were remotely recorded during the test and data was quickly reduced.

a) Radial deformation of the containment wall was measured at 15 locations in the thickened Equipment Hatch boss and the transition area from the thickened boss to the 4'-6" cylinder wall.

b) Diameter change in the containment structure was measured at 10 locations spaced at approximately 10'-0" between elevations 101'-0" and 191'-0".

c) Radial deflection of the containment cylinder wall was measured at elevation 91'-0".

d) Vertical deflection of the Containment was measured at elevations 95'-0", 143'-0" and 191'-0" and at the apex of the dome. Redundancy was provided for the measurement at the apex of the dome.

Detailed crack measurements were made prior to the test, at peak test pressure of 54 psig, and following depressurization at five areas of the exterior shell, each of at least forty square feet in area. The areas of detailed measurement were: a quadrant of the personnel lock concrete boss, and ten foot wide strips spanning elevations 43'-0" to 48'-0", 115'-0" to 120'-0", and 188'-0" to 193'-0".

In addition, the exposed surface of the containment shell was visually inspected prior to the test, at 41 psig during the ILRT, and following depressurization. These inspections were for purposes of monitoring the general crack pattern and for specifically following the behavior of the most significant crack.

5.1.7.3 Acceptability of Testing Program

AEC Safety Guide No. 18 "Structural Acceptance Test for Concrete Primary Reactor Containments" was followed for testing except in the following areas:

1) The pattern of measurement points around the largest opening (equipment hatch) were not as shown in Figure C of Safety Guide 18 which indicated 12 points symmetrically located to measure radial and tangential deflections. The Indian Point 3 Structural Integrity Test required taking of radial measurements at 15 locations around the equipment hatch.

Due to access restrictions, no deflection readings were taken on the lower vertical axis of this opening; the 15 measurement locations were symmetrically positioned in the remaining accessible area around this opening. Tangential deflections were not taken, as they were insignificant compared to the radial deflections. The second largest opening (personnel hatch) was structurally loaded in a manner similar to the equipment hatch; no deflection measurements were taken for the personnel hatch opening. This program of radial deflection measurements provided the necessary data to verify that anticipated deformations were taken into account and were within acceptable limits.

2) The structural integrity of the OEH was tested in the Vendor's shop to 7.5 psig for 10 minutes, then the pressure was dropped to 6 psig and the air supply was closed. All tests were performed in accordance with the requirements of ASME B&VP code Section VIII, 1989 Code Part UG-99 or UG-100 for the fabricated Carbon Steel.

5.1.7.4 Post-Operational Tests

The double penetrations and most weld seam channels which were installed on the inside of the liner in the Containment are continuously pressurized to provide a continuous, sensitive and accurate means of monitoring their status with respect to leakage. Certain liner welds are no longer continuously pressurized. Therefore, the leakage status of these welds is no longer continuously monitored. The integrity of these welds is verified by integrated leak rate testing.

No periodic structural integrity tests of the Containment are planned. Periodic peak pressure containment integrated leakage rate test (ILRTs) are performed in accordance with the Technical Specifications. Peak pressure tests are to be conducted as appropriate in the event major maintenance or major plant modifications are made. As a prerequisite to the ILRT, a detailed visual examination of the accessible interior and exterior surfaces of the containment structure and its components is required to uncover any evidence of deterioration which may affect the containment integrity. However, no degradation of structural integrity is expected. The Authority does not consider periodic structural integrity tests as warranted either separately or in conjunction with other tests.

The Containment Leakage Rate Testing Program details requirements for inspection of the accessible interior and exterior surfaces of the containment structure and its components. This periodic surveillance of the Containment and associated structures is visual and includes critical areas as well as a general examination of the accessible surfaces for deterioration. The inspection is also performed prior to any integrated leak test. The insulation attached to the steel liner is designed so that sections can be removed to facilitate inspection of the liner.

These components are considered ASME Section XI Class MC or CC components and any repair or replacement activities shall be performed in accordance with ASME Section XI Subsections IWE and IWL of the ASME Code, 1992 Edition with certain exceptions whenever specific relief is granted by the NRC.

Provisions have been made for access to the upper external parts of the containment structure. These provisions consider the use of movable scaffolding while performing periodic inspection and testing during the service life of the facility.

References

1. Stellmeyer, J. B., W. H. Munse and E. A. Selby, "Fatigue Tests of Plates and Beams with Stud Shear Connections." Highway Research Record, Number 76.

2. Singleton, Robert C. "The Growth of Stud Welding." Welding Engineer, July 1963.

3. United States Atomic Energy Commission – Nuclear Reactors and Earthquakes, TID-7024, 1963.

4. Blume, J., N. Newark, L. Corning – Design of Multistory Reinforced Concrete Building for Earthquake Motions – Portland Cement Association.

5. Timoshenko, S., and S. Woinowsky-Kreiger, Theory of Plates and Shells, Second Edition, McGraw-Hill, 1954.

6. American Concrete Institute, Code for Reinforced Concrete Chimney Design, ACI-505.

Table 5.2.1 provides a summary of containment isolation valve type, actuator, and closure time established for systems penetrating containment.

With respect to numbers and locations of isolation valves, the criteria applied were generally those outlined by the seven classes described in Section 5.2.2. Specific containment isolation valves are listed in FSAR Table 5.2-3.

5.2.2 System Design

The seven classes listed below are general categories into which line penetrating containment may be classified. The seal water referred to in the listing of categories is provided by the Isolation Valve Seal Water System described in Section 6.5. The following notes apply to these classifications:

1) The "not missile protected" designation refers to lines that are not protected throughout their length inside containment against missiles generated as the result of a Loss-of-Coolant-Accident. These lines, therefore, are not assumed invulnerable to rupture as a result of a Loss-of-Coolant Accident.

2) In order to qualify for containment isolation, valves inside the Containment must be located behind the missile barrier for protection against loss of function following an accident.

3) Manual isolation valves that are locked closed or otherwise closed and under administrative control during power operation qualify as automatic trip valves.

4) A check valve qualifies as an automatic trip valve in certain incoming lines not requiring seal water injection.

5) The double disk type of gate valve was used to isolate certain lines. When sealed by water or gas injection, this valve provides two barriers against leakage of radioactive liquids or containment atmosphere. In certain cases, a double disc valve was used in place of two valves in series having seal water or gas injection between them.

6) In lines isolated by globe valves in series (inboard and outboard) outside containment and provided with seal water injection, the following applies:

   a) On process lines ingressing containment (incoming lines) IVSWS will be required to wet the stem packings on both the inboard and outboard valve. IVSW wets the valve plug as well as the stem packing of the RCP seal water injection line containment isolation valves (CH-MOV-250A through D),

   b) On process lines egressing containment (outgoing lines) IVSWS will be required to wet only the stem packing on the inboard valve. One exception would be the Steam Generator Blowdown CIVs where both the inboard and outboard valves stem packings are wetted by IVSWS.

7) Excessive loss of seal water through an isolation valve that fails to close on signal is prevented by the high resistance of the seal water injection line. A water seal at the failed valve was assured by proper slope of the protected line, or a loop seal, or by additional valves on the side of the isolation valves away from the Containment.

8) Lines penetrating containment were designed to the same seismic criteria as the containment vessel up to and including the second isolation barrier. These portions of the penetrating lines are therefore to be considered extensions of the containment.

A review of the Containment Isolation System (NUREG-0578) indicated that there were a number of valves, which automatically reset to the previous position upon reset of containment Phase A isolation. These valves were under operator control via operating procedures to be placed in the closed position prior to resetting of Phase A. Circuits for these valves have been modified to preclude automatic opening on reset. The modification to the valve circuits entailed the installation of pushbuttons that work in conjunction with the containment isolation reset switches so that each valve control circuit has to be reset or the valve will be inhibited from opening.

Class 1 (Outgoing Lines, Reactor Coolant System)

Outgoing lines connected to the Reactor Coolant System which are normally or intermittently open during reactor operation were provided with at least two automatic trip valves in series located outside the Containment. Automatic seal water injection was provided for line in this classification.

Class 2 (Outgoing Lines)

Outgoing lines not connected to the Reactor Coolant System which are normally or intermittently open during reactor operation, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident, were provided, as a minimum, with two automatic trip valves in series outside containment. Automatic seal water injection was provided for lines in this classification with the exception of the reactor coolant pump seal water return line, which was provided with manual seal water injection. Most of these lines are not vital to plant operation following an accident.

Class 3 (Incoming Lines)

Incoming lines connected to open systems outside containment, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident were provided with one of the following arrangements outside containment:

1) Two automatic trip valves in series, with automatic seal water injection. This arrangement was provided for lines, which are not necessary to plant operation after an accident.

2) Two manual isolation valves in series, with manual seal water injection. This arrangement was provided for lines, which remain in service for a time, or are used periodically, subsequent to an accident.

Incoming lines connected to closed systems outside containment, and not missile protected or which can otherwise communicate with the containment atmosphere following an accident were provided either with two isolation valves in series outside containment with seal water injection between them or, at a minimum with one check valve or normally closed isolation valve located either inside or outside containment.

The closed piping system outside containment provides the necessary isolation redundancy for lines, which contain only one isolation valve.

Exceptions are the containment spray headers and the safety injection header associated with the boron injection tank, which was valved in accordance with safeguards requirements. The containment spray headers have locked-open double disk gate valves while the safety injection header has either single normally-open double disk gate valves or two normally open gate valves arranged in series.

Class 4 (Missile Protected)

Incoming and outgoing lines which penetrate the Containment and which are normally or intermittently open during reactor operation and are connected to closed systems inside the Containment and protected for missiles throughout their length were provided with at least one isolation valve located outside the Containment. Seal water injection was provided for certain lines in this classification.

Class 5 (Normally Closed Lines Penetrating the Containment)

Lines which penetrate the Containment and which can be opened to the containment atmosphere but which are normally closed during reactor operation were provided with two isolation valves in series or one isolation valve and one blind flange.

Class 6 (Special Service)

There are a number of special groups of penetrating lines and containment access openings. Some of these are discussed below.

Each ventilation purge duct penetration was provided with two tight-closing butterfly valves, which are closed during reactor power operation and are actuated to the closed position automatically upon a containment isolation or a containment high radiation signal.

One valve is located inside and one valve is located outside the Containment at each penetration. The space between valves is pressurized by air from the Penetration and Weld Channel Pressurization System, whenever they are closed.

The containment pressure relief line is similarly protected. However, since the line can be opened during reactor power operation, three tight closing butterfly valves in series are provided, one inside and two outside the Containment. These valves also are actuated to the closed position upon a containment isolation or containment high radiation signal. The two intravalve spaces are pressurized by air from the Penetration and Weld Channel Pressurization System whenever they are closed.

The equipment access closure is a bolted, gasketed closure, which is sealed during reactor operation. The personnel air locks consist of two doors in series with mechanical interlocks to assure that one door is closed at all times. Each air lock door and the equipment closure were provided with double gaskets to permit pressurization between the gaskets by the Penetration and Weld Channel Pressurization System, Section 6.6.

The fuel transfer tube penetration inside the Containment was designed to present a missile protected and pressurized double barrier between the containment atmosphere and the atmosphere outside the Containment. The penetration closure was treated in a manner similar to the equipment access hatch. A positive pressure is maintained between the double gaskets to complete the double barrier between the containment atmosphere and the inside of the fuel transfer tube. The interior of the fuel transfer tube is not pressurized. Seal water injection is not required for this penetration.

The following lines would be subjected to pressure in excess of the Isolation Valve Seal Water System design pressure (150 psig) in the event of an accident, due to operation of the recirculation pumps:

1) Residual heat removal loop return line

2) Bypass line from residual heat exchanger outlet to safety injection pumps suction

3) Residual heat removal loop sample line

4) Recirculation pump discharge sample line

5) Residual heat removal pump miniflow line

6) Residual heat removal loop outlet line

Lines 1, 2, and 6 are isolated by double disc gate valves, while line 3, 4 and 5 are each isolated by two valves in series. These valves can be sealed by nitrogen gas from the high pressure nitrogen supply of the Isolation Valve Seal Water System.

A self contained pressure regulator operates to maintain the nitrogen injection pressure slightly higher than the maximum expected line pressure. The nitrogen gas injection is manually initiated.

Lines which communicate with the containment atmosphere at all times (normally filled with air or vapor) include:

1) Steam jet air ejector return line to containment

2) Containment radiation monitor inlet and outlet lines.

In an accident condition, the space between the two containment isolation valves in each line is sealed by pressurizing with air from the Penetration and Weld Channel Pressurization System. The air is introduced into each space above the containment calculated peak accident response pressure through a separate line from the Penetration and Weld Channel Pressurization System. Parallel (redundant) fail open valves in each injection line open on the appropriate containment isolation signal to provide a reliable supply of pressurizing air. A flow limiting orifice in each injection line prevents excessive air consumption if one of these valves spuriously fails open, or if one of the containment isolation valves fails to respond to the "trip" signal.

Class 7 (Steam and Feedwater Lines)

These lines and the shell side of the steam generator are considered basically as an extension of the containment boundary and as such must not be damaged as a consequence of Reactor Coolant System damage. This required that the steam generator shell, feed and steam lines within the Containment be classified and designed for the Reactor Coolant System missile-protected category. The reverse is also true in that a steam line break is not to cause damage to the Reactor Coolant System.

5.2.2.1 Isolation Valves and Instrumentation Diagrams

Figures 5.2-1 through 5.2.28 show all valves in lines leading to the atmosphere or to closed systems on both sides of the containment barrier, valve actuation and preferential failure modes, the application of "trip" (containment isolation) signals, relative location of the valves with respect to missile barriers, and the boundaries of seismic Class I designed lines. Figure 5.2-29 defines the nomenclature and symbols used. Individual containment isolation valves are listed in Section 5.2 of the FSAR and Table 5.2-3.

5.2.2.2 Normally Closed Isolation Valves

Table 5.2-3 identifies those isolation valves which are either locked closed, or normally closed, (under administrative control) in normal position and relates to Figures 5.2.1 through 5.2-29.

5.2.2.3 Valve Parameters Tabulation

A summary of the fluid systems lines penetrating containment and the valves and closed systems employed for containment isolation is presented in Table 5.2-3. Each valve is described as to type, operator, position indication and open or closed status during normal operation, shutdown and accident conditions. Information is also presented on valve preferential failure mode, automatic trip by the containment isolation signal, and the fluid carried by the line.

Containment isolation valves were provided with actuation and control equipment appropriate to the valve type. For example, air operated globe and diaphragm (Saunders Patent) valves are generally equipped with air diaphragm operators, with fail-safe operation provided by the control devices in the instrument air supply to the valve. Motor operated gate valves are capable of being supplied from reliable onsite emergency power as well as their normal power source. Manual and check valves, of course, do not require actuation or control systems.

The automatically tripped isolation valves are actuated to the closed position by one of two separate containment isolation signals. The first of these signals is derived in conjunction with automatic safety injection actuation, and trips the majority of the automatic isolation valves. These are valves in the so-called "non-essential"** process lines penetrating the containment. This is defined as a "Phase A" isolation and the trip valves are designated by the letter "T" in the isolation diagrams, Figures 5.2-1 through 5.2-29. This signal also initiates automatic seal water injection (See Section 6.5). The second, or "Phase B," containment isolation signal is derived upon actuation of the Containment Spray System, and trips the automatic isolation valves in the so called "essential"* process lines penetrating the containment. These trip valves are designated by the letter "P" in the isolation diagrams.

* "Essential" are those lines required to mitigate an accident, or which, if unavailable, could increase the magnitude of the event. Also, those lines which, if available, would be used in the short term (24 to 36 hours) to restore the plant to normal operation following an event which has resulted in containment isolation.

** "Non-Essential" are those lines which are not required to mitigate or limit an accident, which if required at all would be required for long-term recovery only, i.e., days or weeks following an accident.

Ch. 6

The motor operated valves in the recirculation suction lines from the containment sump are maintained in the normally closed position at all times, however, they could be opened to allow for residual heat removal pump recirculation operation if that mode was required.

The valves are exercised in accordance with Technical Specification requirements. The valves are operated one at a time and each valve is returned to its normal position before exercising the next one.

No automatic opening features are provided; hence, the probability of a spurious signal to open the valves is nil. The only time these valves are opened is for periodic testing and the procedure ensures that both valves are closed immediately after the test. In addition, the two valves are provided in series to protect against the inadvertent opening of one valve.

The procedure used for periodic testing of these valves ensures that the only water which would be drained from these lines is the small amount trapped between the two valves. This water will discharge to the containment sump. The sump contains two sump pumps which operate on level control and will periodically pump the sump contents to the waste holdup tank during normal plant operation.

For small breaks the depressurization of the Reactor Coolant System is augmented by steam dump and auxiliary feed water addition to the Steam System. For the small breaks in the Reactor Coolant System where recirculated water must be injected against higher pressures for long term core cooling, the system is arranged to deliver the water from the residual heat exchangers to the high-head safety injection pump suction and, by this external recirculation route, to the reactor coolant loops. Thus, if depressurization of the Reactor Coolant System proceeds slowly, the safety injection pumps may be used to augment the flow-pressure capacity of the recirculation pumps in returning the spilled coolant to the reactor.

The recirculation pumps, the residual heat exchangers, piping and valves vital to the function of the recirculation loop are located in a missile-shielded space inside the polar crane support wall on the west side of the reactor primary shield.

There are two recirculation related sumps within the Containment, the recirculation sump and the containment sump. Both sumps collect liquids discharged into the Containment during the injection phase of the design basis accident.

The recirculation sump contains two screens through which the recirculated water must flow before entering the pumps. The first screen consists of a floor grating (1" x 4") which covers the sump on the basement floor. The purpose of the grating is to prevent large particles form entering the sump. The second screen is located in the sump and has the capability to exclude particles greater than 1/8 inch in diameter from the recirculation pump suction. This floor grating has a total surface area of 48.3 ft^2. Since all recirculated water passes through both screens before entering the pumps, particles in excess of 1/8 inch diameter are precluded from entering these lines. The water velocity through the sump is less than one foot per second.

The containment sump contains two screens for the purpose of preventing particles greater than 1/8 inch diameter from entering the residual heat removal pump suction. The first screen consists of 1" x 4" floor grating with an area of 41.3 ft^2; the second screen is located in the sump. The water velocity through the sump is less than one foot/second.

The low head external recirculation loop via the containment sump line and the residual heat removal pumps provides backup recirculation capability to the low head internal recirculation loop. The containment sump line is contained within a concentric guard pipe which is connected to the containment liner and terminates within a leak tight compartment. This sump line has two remote motor operated normally closed valves for containment isolation purposes, one of which is within this leak tight compartment.

The high head external recirculation flow path via the high head safety injection pumps is only required for the range of small break sizes for which the Reactor Coolant System pressure remains in excess of the shutoff head of the recirculation pumps (or residual heat removal pumps) at the end of the injection phase or to provide hot leg flow during hot leg recirculation.

The external recirculation flow paths within the Primary Auxiliary Building are designed so that external recirculation can be initiated immediately after the accident. Those portions of the Safety Injection System located outside of the Containment which are designed to circulate under post-accident conditions radioactively contaminated water collected in the Containment meet the following requirements:

1) Shielding to maintain radiation levels within the guidelines set forth in 10 CFR 100

2) Collection of discharges from pressure relieving devices into closed systems

3) Means to detect and control radioactivity leakage into the environs to the limits consistent with guidelines set forth in 10 CFR 100.

This criterion is met by minimizing leakage from the system. External recirculation loop leakage is discussed in Section 6.2.3.

One pump (either recirculation or residual heat removal) and one residual heat exchanger of the recirculation system provides sufficient cooled recirculated water to keep the core flooded with water by injection through the cold leg connections while simultaneously providing, if required, sufficient containment spray flow to prevent the containment pressure from rising above design limits because of the boiloff from the core. Only one pump and one heat exchanger are required to operate for this capability at the earliest time recirculation is initiated. The design ensures that heat removal from the core and Containment is effective in the event of a pipe or valve body rupture.

<u>Cooling Water</u>

The Service Water System (Section 9.6.1) provides cooling water to the component cooling loop, which in turn, cools the residual heat exchangers, all of which are part of the Auxiliary Cooling Systems (Section 9.3). Three conventional service water pumps are available to take suction from the river and discharge to the two component cooling heat exchangers. Three component cooling pumps are available to discharge through their heat exchangers and deliver to the two residual heat exchangers. With the component cooling water system in long term recirculation mode, the following components are required in order to meet core cooling requirements, one residual heat removal pump and heat exchanger, one component cooling water pump, one component cooling water heat exchanger, one service

water pump on the nonessential header, and two essential service water pumps on the essential header. All of this equipment with the exception of the residual heat exchangers is located outside Containment.

Containment Building Water Level Monitoring

Continuous indication of containment water level during and after an accident is provided by three systems with redundant measuring loops distributed as follows:

1) Containment Sump (El. 38' 3"), narrow range, 0' to 10' of water.

2) Recirculation Sump (El, 34' 0"), narrow range, 0' to 14' of water.

3) Containment Building (El, 46' 0"), wide range, 0' to 8' of water.

Each loop consists of a sensor and a transmitter located inside the containment building, a recorder and power supply at the control room. Refer to Figure No. 6.2-1A.

Change-Over from Injection Phase to Recirculation Phase

Assuming that the three high head safety injection pumps, the two residual heat removal pumps, and the two containment spray pumps (Section 6.3) are running at their maximum capacity, the time sequence for the changeover from injection to recirculation in the case of a large rupture beginning form the time of the safety injection signal:

1) In approximately ten minutes, sufficient water has been delivered to provide the required NPSH to start the recirculation pumps.

2) In approximately 15 to 20 minutes, (1) one of two low level alarms on the RWST sounds, and the redundant containment recirculation sump level indicators show the sump water level. The alarm serves to alert the operator to start the switchover to the recirculation mode. The redundant containment recirculation sump level indicators provide verification the RWST water has been delivered during the injection phase, in addition to providing consideration to the case of a spurious (i.e., early) RWST low level alarm. The operator would see on the control board that the redundant recirculation sump level indications are at the appropriate points; switch-over to the recirculation phase of safety injection is performed at this time.

3) With the initiation of the switch sequence (e.g., Switch No. 1), only one spray pump will continue in operation. This spray pump will continue to draw from the RWST for approximately 25 minutes to assure that the contents of the spray additive tank have been completely mixed with the spray liquid.

Recirculation pump motors are 2'-2" above the highest water level after addition of the injected water to the spilled coolant.

The entire switchover from injection to recirculation phase is carried out by manually initiating equipment starts/stops and closing a series of switches (each of which carries out several operations) located in the Control Room. At only two points in the switchover routine is any reliance on instrumentation necessary:

1) When the low level alarm occurs from the refueling water storage tank low level (LT-920 and/or LIC-921 outside containment), the operator is alerted to begin closing the series of recirculation switches. The low level alarm for each instrument is set to actuate when there is between 10.5 feet and 12.5 feet of water in the tank.

2) After closing switch 4, the operator is required to make a decision whether to close switch 5 or switch 6. The basis for this decision is the flow reading on the flow meters FT-946A, 946B, 946C and 946D. If three or more of these flow meters each indicate greater than zero and the two lowest of these readings are at least 350 gpm, or, with two flowmeters reading zero and the flow reading on each of the other two flowmeters is at least 610 gpm, the operator will close switch 6; otherwise, the operator will close switch 5.

Analysis indicates that approximately 555 gpm to the core is required to match boil-off at 1398 seconds (the earliest time at which recirculation could be initiated). This includes a 9% penalty to allow for the effects of hot metal quenching. The flow rates that follow ensure an actual flow of ≥555 gpm. Accordingly, a requirement of 610 gpm minimum flow rate on one loop (or 350 gpm on each of the two loops) has been specified to account for uncertainties in flow measurement and to provide margin.

The decision making process with regard to the flow to the Reactor Coolant System via the low head injection lines is based on readings of the four flowmeters. The rationale for this basis is the following:

For four flow meters each reading greater than zero (i.e., none indicating zero flow):

1) Assume one flow meter fails to an inaccurate reading of any value, either high or low (as a result of a single failure); for example, if the flow rate reads greater than 350 gpm but delivery is unknown, assume zero flow.

2) Of the three remaining flow meters, assume the highest reading meter is connected to the spilling line; therefore, flow is ineffective.

3) For the two remaining flow meters, their total flow is delivered to core; if at least 350 gpm is indicated, flow requirements are satisfied using low head recirculation.

Refueling Water Storage Tank

In addition to its normal duty to supply borated water to the refueling canal for refueling operations, this tank provides borated water to the safety injection pumps, the residual heat removal pumps and the containment spray pumps for the Loss-of-Coolant Accident. During plant operation, it is aligned to these pumps.

The capacity of the refueling water storage tank is based on the requirement for filling the refueling canal. When filled to Technical Specification requirements, approximately 342,200 gallons is available for delivery. One low level alarm is set to actuate at between 10.5 feet and 12.5 feet of water in the tank. This tank capacity and these alarm settings provide an amount of borated water to assure:

1) A sufficient volume of water on the floor to permit the initiation of recirculation (195,800 gal).

2) A volume sufficient to allow switchover to recirculation pumps, containment pressure relief, and sump pH control via containment spray system following a reactor coolant pressure boundary break (66,700 gal).

3) Adequate volume to allow for instrument uncertainties (total 78,000 gal for all instruments).

4) The total RWST volume, when added with accumulator discharge to the reactor coolant system, will assure no return to criticality with the reactor at cold shutdown and no control rods inserted into the core.

The water in the tank is borated to a concentration which assures reactor shutdown by at least 5% $\Delta k/k$ when all RCC assemblies are inserted and when the reactor is cooled down for refueling. The maximum boric acid concentration is approximately 1.5 weight percent boric acid. At 32°F the solubility limit of boric acid is 2.2%. Therefore, the concentration of boric acid in the refueling water storage tank is well below the solubility limit at 32°F.

The contents of the Refueling Water Storage Tank are kept above 32°F by a steam heated, austenitic stainless steel pipe coil in the bottom of the tank. Steam is supplied to this coil through a single header from the auxiliary boilers which are used to supply all required auxiliary steam to Indian Point 3.

The passive heating coil and passive single supply header are supplied with steam from any one of five sources. In the remote case of loss of steam to this tank, there would be a time period of at least 24 hours available for repair or connection to another steam source before freezing problems would arise, even under the most severe weather conditions. If the electrical heat tracing on the tank discharge line remains operable it is very probable that a freezing problem would not arise.

The steam to the heating coil is automatically flow controlled to maintain a minimum tank water temperature of 35°F. It is also controlled automatically by pressure to maintain a maximum steam pressure in the coil of 4 to 5 psig. Thus, when the tank is normally filled with borated water, the water pressure outside the heating coil will be approximately 15 psig, preventing leakage of steam out of the coil and subsequent dilution of the borated water.

All outdoor piping connected to the Refueling Water Storage Tank is electrically heat traced. The failure of any section of heat tracing is annunciated in the Control Room. The power source for the heat tracing can be manually switched between two MCC's, each powered automatically by different emergency diesel generator buses.

The design parameters are presented in Table 6.2-4

Pumps

Class I (seismic) pumps in the Emergency Safeguards Systems, their required Net Positive Suction Head (NPSH) at extreme operating conditions, the fluid operating temperature, the NPSH available, the atmospheric pressure assumption, and the elevation of each pump are given in Table 6.2-13.

The Internal Recirculation Pump NPSH data in Table 6.2-13 are given for a single pump operation, which represents the most limiting case for maximizing flow and NPSHR. As shown on the Figure 6.2-4 pump curve, an NPSHR of 12.7 ft is required for a flow of 3530 gpm. This flow rate has been increased to account for pump recirculation and the measurement uncertainty associated with procedural instructions to limit pump flow to no greater than 3000 gpm while feeding the core and the recirculation spray header. These pumps were designed to operated under cavitating conditions. The pump vendor has confirmed that reduced levels of NPSH are acceptable (Reference 7), such that the Recirculation Pumps can operate indefinitely with an NPSH value @90% of that required on pump curve (Figure 6.2-4). The effective limit for NPSHR thus becomes 11.4 ft. at an indicated flow of 3000 gpm, or an actual flow of 3530 gpm at the pump nozzles. See additional discussion of Recirculation Pump NPSH in Section 6.2.3.

An analysis predicts that, for the large break LOCA, there will be 11.76 ft. of NPSH available, which credits the remainder of the RWST water delivered to the containment prior to the start of recirculation containment spray (References 8, 9). In the case of a small-break LOCA, when elevated RCS pressure would preclude direct low head recirculation, high head recirculation would then be established using the Recirculation Pump(s) to deliver a suction supply to the SI Pumps. During high head recirculation, the Recirculation Pumps operate at lower flow rates and the NPSH requirements are correspondingly lower.

NPSH calculations assume saturated water in the sumps so that no credit is taken for containment pressure exceeding the vapor pressure of the sump water. While this conservative assumption is appropriate at accident initiation, it does not allow any credit for the increase of NPSHA which would result from the gradual cooling of the sump fluid to below saturated conditions.

A review performed pursuant to NRC Generic Letter 85-22 had also established that the actual containment water level would be well above the minimum switchover level indicated in Table 6.2-13. This actual water level would provide sufficient additional NPSH available to overcome the head loss effects of debris which has been postulated to result from the destruction of steam generator thermal insulation by LOCA jet forces.

The three (high head) safety injection pumps for supplying borated water to the Reactor Coolant System are horizontal centrifugal pumps driven by electric motors. Parts of the pump in contact with borated water are stainless steel or equivalent corrosion resistant material. A minimum flow bypass line is provided on each pump discharge to recirculate flow to the refueling water storage tank in the event the pumps are started with the normal flow paths blocked. The bypass line joins a common miniflow line shared by the other pumps. Each safety injection pump is sized at 50% of the capacity required to meet the design criteria outlined in Section 6.2.1. The design parameters are presented in Table 6.2-5, and Figure 6.2-2 gives the performance characteristics of these pumps.

Operation of this system with three of the four available accumulators delivering their contents to the reactor vessel (one accumulator spilling through the break) prevents fuel clad melting and limits metal-water reaction to an insignificant amount (less than 1%).

The function of the safety injection or residual heat removal pumps is to complete the refill of the vessel and ultimately return the core to a sub-cooled state. The flow from either two safety injection pumps or one residual heat removal pump is sufficient to complete the refill with no loss of level in the core.

The design features applied to the Residual Heat Removal System (RHRS) Valves 730 and 731, that isolate it from the Reactor Coolant System provide a diverse combination of control interlock and mechanical limitations preventing improper opening of these valves and also pressure relief capacity capable of limiting pressure if the valves are not closed upon startup of the plant. These features are:

1) That the valves that are separately interlocked with independent pressure control signals to prevent their being opened whenever the Reactor Coolant System pressure is greater than a designated setpoint (which is below the RHRS design pressure).

   The pressure interlock was not specifically designed to meet the requirements of IEEE Standard 279-1971. However, each valve, its associated pressure channel and related circuitry are powered from separate instrument buses, and wiring separation is provided to preclude any single failure from rendering both of the valves' control circuits inoperable. Each of the pressure channels is provided with separate Control Room indication to show channel operability.

   A separate pressure interlock is provided for each of the two Valves Nos. 730 and 731. Each pressure interlock prevents its valve from being opened when the Reactor Coolant System pressure is greater than a designated open permissive setpoint and also automatically closes the valve whenever the Reactor Coolant System pressure is above a designated auto-close setpoint. These setpoints are below the design pressure of the RHRS.

   While the automatic closure interlock for MOV-730 and –731 will prevent over-pressurizing the RHR system piping during an RCS pressure increase transient, this interlock will isolate the suction source of the operating RHR pump(s), potentially causing pump failure. In order to prevent inadvertent isolation of the RHR pump suction, this auto-closure interlock may be defeated by de-energizing the motor operators to MOV-730 and –731. Prior to de-energizing these MOV's Reactor Coolant System T_{ave} must be below 200°F, depressurized and vented through a minimum equivalent opening of two (2) square inches.

2) That the Reactor Coolant System pressure interlocks meet single failure criteria.

3) That the motors are qualified in accordance with IEEE 323-1974, IEEE 344-1975, IEEE 382-1972 for increased reliability and operability in the normal and accident containment environment.

The Residual Heat Removal System was designed for a pressure of 600 psig and 400°F and was hydrostatically tested at a pressure of 900 psig prior to initial operation. Insofar as the piping itself is concerned, the piping code (USAS B31.1) allows a rating of 700 psig at 400°F for schedule 40 stainless steel pipe. Thus the piping system, as presently designed, incorporates a considerable margin in that it is rated at a pressure-temperature condition which is less than that allowed by Code. It is also noted that the Code allows an overpressure allowance above the design pressure under transient conditions.

4) That the RHRS is equipped with a pressure relief valve RV-1836 sized with a relief capacity of 400 gpm. This is a diverse backup to administrative closure of the isolation valves prior to startup to prevent overpressurization when returning the plant to operation. In addition, Technical Specification Section 3.4.12 restricts operation of the SI pumps when the RCS average cold leg temperature is below the OPS enable temperature. These restrictions help to preclude RHR overpressurization.

5) To preclude spurious closure of the valves, the control circuitry is of the energize-to-actuate principle.

6) That each of the pressure channels has a separate Control Room indication to show channel operability.

7) Open/close position indication lights are provided for these valves as well as a visual and audible alarm to indicate when either valve leaves its full open position.

Initial response of the injection systems is automatic with appropriate allowances for delays in actuation of circuitry and active components. The active portions of the injection systems are automatically actuated by the safety injection signal (Chapter 7). In addition, manual actuation of the entire injection system and individual components can be accomplished from the Control Room. In analysis of system performance, delays in reaching the programmed trip points and in actuation of components are conservatively established on the basis that only emergency onsite power is available.

The starting sequence of the safety injection and residual heat removal pumps and the related emergency power equipment is designed so that delivery of the full rated flow is reached within 27 seconds after the process parameters reach the set points for the injection signal.

EVENT	SECONDS
Time to initiate the safety injection signal	2
Time for diesel generators to come up to speed	10
Time for safety injection pumps to come up to speed	10
Time for Residual Heat Removal Pumps to come up to speed	5
Total	27

Motor control centers are energized and injection valves are opened during this time to allow pumped ECCS delivery.

Recirculation Pumps

The NPSH for the recirculation pumps is evaluated for recirculation operation. The NPSH available is determined from the elevation head of the water above the pump inlet in the sump.

The internal recirculation pumps are conventional vertical condensate pumps which in the past have been used with NPSH control. This type control used the NPSH available in the condenser hot well to control the discharge condition of the pump thus resulting in continuous pump cavitation. No approach to cavitating conditions are anticipated for the normal case of two pumps operating. If, however, only one pump is delivering through two heat exchangers with full saturated fluid, the operator is advised to throttle back via the heat exchanger butterfly valves to avoid long term operation at or near cavitating conditions. For cases in which a single recirculation pump is providing flow for both core cooling and containment spray, the pump may be expected to operate in a cavitating mode while recirculation spray is in service.

6.2.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system when in MODES 1, 2, 3, and 4.

6.2.5 Inspections and Tests

Inspection

All components of the Safety Injection System are inspected periodically to demonstrate system readiness.

The pressure containing components are inspected for leaks from pump seals, valve packing, flanged joints and safety valves during system testing.

In addition, to the extent practical, the critical parts of the reactor vessel internals, pipes, valves and pumps are inspected visually or by boroscopic examination for erosion, corrosion, and vibration wear evidence and for non-destructive test inspection where such techniques are desirable and appropriate.

Pre-Operational Testing

Component Testing

Pre-operational performance tests of the components were performed in the manufacturer's shop. The pressure-containing parts of the pump were hydrostatically tested in accordance with Paragraph UG-99 of Section VIII of the ASME Code. Each pump was given a complete shop performance test in accordance with Hydraulic Institute Standards. The pumps were run at design flow and head, shutoff head and at additional points to verify performance characteristics. NPSH was established at design flow by means of adjusting suction pressure for a representative pump.

The remote operated valves in the Safety Injection System are motor operated. Shop tests for each valve included a hydrostatic pressure test, leakage tests, a check of opening and closing time, and verification of torque switch and limit switch settings. The ability of the motor operator to move the valve with the design differential pressure across the gate was demonstrated by opening the valve with an appropriate hydrostatic pressure on one side of the valve.

The recirculation piping and accumulators were initially hydrostatically tested at 150 percent of design pressure.

The service water and component cooling water pumps were tested prior to initial operation.

System Testing

Initial functional tests of the core cooling portion of the Safety Injection System were conducted before initial plant startup. These tests were performed following the flushing and hydrostatic testing of the system and with the Reactor Coolant System cold. The Safety Injection System valving was set initially to simulate the system alignment for Plant Power Operation.

The functional tests were divided into two parts:

1) Demonstrating the proper function of instrumentation and actuation circuits, confirm valve operating times, confirm pump motor starting times, and demonstrate the proper automatic sequencing of load addition to the emergency diesels.

   These tests were repeated for the various modes of operation needed to demonstrate performance at partial effectiveness, i.e., to demonstrate the proper loading sequence with two of the three emergency diesels, and to demonstrate the correct automatic starting of a second pump should the first pump fail to respond. These tests were performed without delivery of water to the Reactor Coolant System, but included the starting of all pumping equipment involved in each test.

2) Demonstrating the proper delivery rates of injection water to the Reactor Coolant System.

To initiate the first part of the test, the safety injection block switch was moved to the unblock position to provide control power allowing the automatic actuation of the safety injection relays from the low water level and low-pressure signals from the pressurizer instrumentation. Simultaneously, the breakers supplying outside power to the 480 volt buses were tripped manually and operation of the emergency diesel system automatically commences. The high-head safety injection pumps and the residual heat removal pumps were started automatically following the prescribed diesel loading sequence. The valves were operated automatically to align the flow path for injection into the Reactor Coolant System.

The second portion of the test was initiated by manually starting individual pumps on mini-flow and manually opening the appropriate isolation valve to deliver water to the Reactor Coolant System. Data was taken to verify proper pump performance and flow delivery rates.

The systems were accepted only after demonstration of proper actuation and after demonstration of flow delivery and shutoff head within design requirements.

Post-Operational Testing

Component Testing

Routine periodic testing of the Safety Injection System components and all necessary support systems at power is performed. The safety injection and residual heat removal pumps are to be tested in accordance with the Indian Point 3 Inservice Testing Program, to check the operation of the starting circuits, verify the pumps are in satisfactory running order, and verification is made that required discharge head is attained. No inflow to the Reactor Coolant System occurs whenever the reactor coolant pressure is above 1500 psi. If testing indicates a need for corrective maintenance, the redundancy of equipment in these systems permits such maintenance to be performed without shutting down or reducing load under conditions defined in the Technical Specifications. These conditions include the period within which the component should be restored to service.

The operation of the remote stop valves in the accumulator tank discharge line may be tested by opening the remote test valves just downstream of the stop valve. Flow through the test line can be observed on instruments and the opening and closing of the discharge line stop valves can be sensed on this instrumentation. Test circuits are provided to periodically examine the leakage back through the check valves and to ascertain that these valves seat whenever the Reactor Coolant System pressure is raised.

This test is routinely performed when the reactor is being returned to power after an outage and the reactor pressure is raised above the accumulator pressure. If leakage through a check valve should become excessive, the isolation valve would be closed (the safety injection actuation signal will cause this valve to open should it be in the closed position at the time of a Loss-of-Coolant Accident). The performance of the check valves has been carefully studied and it is concluded that it is highly unlikely that the accumulator lines would have to be closed because of leakage.

The recirculation pumps are normally in a dry sump. Minimum flow testing of these pumps can be performed during refueling operations by filling the recirculation sump and opening the mini flow valve on the discharge of the pump and directing the flow back to the sump. Those service water and component cooling pumps which are not running during normal operation may be tested by alternating with the operating pumps.

The content of the accumulators, the Boron Injection Tank and the Refueling Water Storage Tank are sampled periodically to determine that the required boron concentration is present.

System Testing

System testing can be conducted during plant shutdown to demonstrate proper automatic operation of the Safety Injection System. A test signal is applied to initiate automatic action and verification made that the components receive the safety injection signal in the proper sequence. The Safety Injection and Residual Heat Removal pumps are blocked from starting. Isolation valves in the injection lines are blocked closed so that flow is not introduced into the reactor coolant system. The system test demonstrates the operation of the valves, pump circuit breakers, and automatic circuitry. The test is considered satisfactory if control board indication and visual observations indicate all components have operated and sequenced properly. A complete system test cannot be performed when the reactor is operating because a safety injection signal would cause a reactor trip. The method of assuring complete operability of the Safety Injection System is to combine the system test performed during plant shutdown with more frequent component tests, which can be performed during reactor operation.

The accumulators and the injection piping up to the final isolation valve are maintained full of borated water while the plant is in operation. The accumulators and the high head injection lines are refilled with borated water as required by using the safety injection pumps to recirculate refueling water through the injection lines. A small test line is provided for the purpose in each injection header.

Flow in each of the safety injection headers and in the main flow line for the residual heat removal pumps is monitored by a local flow indicator. Pressure instrumentation is also provided for the main flow paths of the safety injection and residual heat removal pumps. Accumulator isolation valves are blocked closed for this test.

The eight-switch sequence for recirculation operation may be tested following the above injection phase test to demonstrate proper sequencing of valves and pumps. The recirculation pumps are blocked from starting during this test.

The external recirculation flow paths are hydrotested during periodic retests at the operating pressures. This is accomplished by running each pump which could be utilized during external recirculation (safety injection and residual heat removal pumps) in turn at near shutoff head conditions and checking the discharge and recirculation test lines. The suction lines are tested by running the residual heat removal pumps and opening the flow path to the safety injection pumps in the same manner as described above.

During the above test, all system joints, valve packings, pump seals, leak-off connections or other potential points of leakage are visually examined. Valve gland packing, pump seals and flanges are adjusted or replaced as required to reduce the leakage to acceptable proportions. For power operated valves, final packing adjustments are made, and the valves are put through an operating cycle before a final leakage examination is made.

The entire recirculation loop except the recirculation line to the residual heat removal pumps is pressurized during periodic testing of the Engineered-Safety Features components. The recirculation line to the residual heat removal pump is capable of being hydrotested during plant shutdown, and it is also leak tested at the time of the periodic retests of the Containment.

Reliance on Interconnected Systems

For the injection phase, the Containment Spray System operates independently of other Engineered Safety Features following a Loss-of-Coolant Accident, except that it shares the source of water in the Refueling Water Storage Tank with the Safety Injection System. The system acts as a backup to the Containment Air Recirculation Cooling and Filtration System for both the cooling and iodine removal functions. For extended operation in the recirculation mode, water is supplied through recirculation pumps.

During the recirculation phase, some of the flow leaving the residual heat exchangers may be diverted to the containment spray headers or the high head safety injection pumps. Minimum flow requirements are set for the flow being sent to the core and for the flow being sent to the containment spray headers such that at least 555 gpm is sent to the core. Sufficient flow instrumentation is provided so that the operator can perform appropriate flow adjustments with the remote throttle valves in the flow path as shown in Figure 6.2-1B.

Normal and emergency power supply requirements are discussed in Chapter 8.

Shared Function Evaluation

Table 6.3-5 is an evaluation of the main components which have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

Containment Spray Pump NPSH Requirements

The NPSH for the containment spray pumps is evaluated for injection operation. The beginning fill-up period of the injection phase gives the limiting NPSH requirements. The NPSH available is determined from the elevation head and vapor pressure of the water in the RWST and the pressure drop in the piping to the pump. Sufficient NPSH margin is available to prevent cavitation of the CS pumps under all operating conditions.

6.4.3 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system when the reactor is above MODE 5.

6.3.5 Inspections and Tests

Inspections

All components of the Containment Spray System are inspected periodically to demonstrate system readiness.

The pressure containing systems are inspected for leaks from pump seals, valve packing, flanged joints and safety valves during system testing. During the operational testing of the containment spray pumps, the portions of the system subjected to pump pressure are inspected for leaks.

Pre-Operational Testing

Offsite Work

These components in the system were subjected to offsite test work:

a) Spray pumps

b) Spray nozzles

c) Eductors

The spray pumps were subjected to conventional acceptance tests and the performance characteristics plotted to illustrate that the pumps met the design specification.

As part of the development work in support of Westinghouse plant equipment, a nozzle of the type used in the spray system was subjected to a performance test to demonstrate and prove the nozzle characteristics (e.g., flow/pressure drop, droplet size, spread of spray, etc.).

As part of the quality assurance program, a random 25% of the nozzles installed at the Indian Point 3 site were given a general performance test.

The eductors were produced and tested in two stages.

a) A prototype was made to check nozzle calculations prior to manufacture of the stainless steel units

b) A performance test was made by the manufacturer on one of the finished stainless steel units to confirm the capacity at the specified conditions. A sugar-water solution was used to simulate the 30% sodium hydroxide suction fluid.

Onsite Test Work

The aim of onsite testing was to:

a) Demonstrate and prove that the system is adequate to meet the design pressure conditions; outside the Containment this involved part radiographic inspection and part hydro-testing; inside the Containment the spray headers were subjected to 100% radiographic inspection

b) Demonstrate that the spray nozzles in the containment spray header are clear of obstructions by passing air through the test connections

Shared Function Evaluation

Table 6.4-2 is an evaluation of the main components which have been discussed previously and a brief description of how each component functions during normal operation and during the accident.

Reliability Evaluation of the Fan-Cooler Motor

The basic design of the motor and heat exchanger as described herein is such that the incident environment is prevented, in any major sense, from entering the motor winding. When entering in a very limited amount (equalizing motor interior pressure), the incoming atmosphere is directed to the heat exchanger coils where moisture is condensed out. If some quantity of moisture should pass through the coil, the changed motor interior environment would "clean up" as that interior air continually recirculates through the heat exchanger.

It will be noted that the motor insulation hot spot temperature is not expected to exceed 127 C even under incident conditions. Normal life could be expected with a continuous hot spot of 155 C.

During the lifetime of the plant, these motors perform the normal heat removal service and as such are only loaded to approximately 90-100 hp, which is less than half the rated horsepower.

The bearings were designed to perform in the incident ambient temperature conditions. However, it should be noted that the interior bearing housing details are cooled by the heat exchanger. It is expected that bearing temperatures would be 125 C to 140 C under incident conditions.

The insulation has high resistance to moisture and tests performed indicate the insulation system would survive the incident ambient moisture condition without failure (see Appendix 6F). The heat exchanger system of preventing moisture from reaching the winding keeps the winding in much more favorable conditions. In addition, it should be noted that at the time of the postulated incident, the load on the fan motor would increase, internal motor temperature would increase, and would, therefore, tend to drive any moisture if present, out of the winding. Additionally, the motors are furnished with insulation voltage margin beyond the operating voltage of 440 volts.

Following the incident rise in pressure, it is not expected that there will be significant mixing of the motor (closed system) environment and the containment ambient.

The heat exchanger was designed using a conservative 0.001 fouling factor.

To prove the effectiveness of the heat exchanger in inhibiting large quantities of the steam air mixture from impringing on the winding and bearings, a full scale motor of the exact same type as described was subjected to prolonged exposure of accident conditions, which included high pressure and temperature, 100% relative humidity, and chemical spray. The test exposed the motor to a steam air mixture as well as boric acid and alkaline spray at approximately 80 psig and saturated temperature conditions.

Insulation resistance, winding and bearing temperature, relative humidity, voltage and current as well as heat exchanger water temperature and flow were recorded periodically during the test.

Following the test, the motor was disassembled and inspected to further assure that the unit performed as designed. The post-testing inspection showed no degradation of the motor components.

Carbon Filter Performance

The design flow rate through each carbon filter bank is 8000 cfm, at a face velocity of approximately 50 fpm. The bed thickness of 2 inches provides a superficial residence time of 0.2 sec. Under the design conditions of temperature, pressure, and humidity, and with moisture uptake limited to less than 1 gram of water per gram of dry charcoal, the expected penetration of incident 1_2 vapor is less than 0.1%.

An evaluation of the effectiveness of charcoal filters in removing organic iodide from the containment atmosphere is presented in Appendix 6C.

6.4.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the air recirculation units when in MODES 1, 2, 3, and 4.

6.4.5 Inspection and Testing

Inspection

Access is available for visual inspection of the containment fan coolers and recirculation filtration components, including fans, cooling coils, dampers, filter units and ductwork. Provision was made for ready removal of the filters for inspection and testing.

Testing

Component Testing

The HEPA filters used in the containment fan cooler system were specified to operate in the post-accident containment environment. Each filter was subjected to standard manufacturer's efficiency and production tests prior to shipment.

These included flow resistance tests and the Standard Efficiency Penetration Test requiring that penetration does not exceed 0.03 percent for 0.3 micron diameter homogeneous diocylphathalate (DOP) particles.

Evaluation tests were performed on sample filters constructed from the filter medium to demonstrate retention of strength under wet conditions, and to demonstrate the effectiveness of the moisture separator for protecting the HEPA filter as follows:

1) The filter was exposed to a flow of wet steam (at 280 F, 50 psig, and 100% R.H.) and water spray (with 2500 ppm boron, pH of 10) in a test facility which simulated the actual filter installation. The water was injected ahead of the filter with a nozzle designed to produce a fine spray. Free (unentrained) moisture was removed by means of a moisture separator upstream of the HEPA filter but no provisions were made for removal of entrained moisture entering the HEPA filter.

2) The filter pressure drop was measured to demonstrate that its resistance to flow under the simulated accident conditions did not significantly increase.

Only filters of a type which have been certified to have passed these tests were accepted for initial use or replacement in the fan coolers application.

Any of the activated carbon filter adsorbers in the air handling units can be removed and tested periodically for effectiveness in removing methyl iodine forms. In addition, periodic in-place testing of the filtration assemblies is made by injection of a DOP aerosol in the air stream at the filter inlet to verify the leak-tightness of individual filter elements and their frame seals. The activated charcoal used shall have an ignition temperature not less than 300 °F.

The in-place testing of HEPA filters with DOP aerosol is performed to demonstrate gasket and media integrity, and overall bank efficiency, rather than an investigation of individual pinhole leaks in the filter media. Test procedures are available at the plant site for inspection.

Large filter installations are tested to within 20% of the full rated flow. Besides limiting the quantity of DOP to be introduced into the ventilation system and containment, this is the flow rate at which filter imperfections would most readily be noticeable. At higher flow rates, the turbulent flow through pinhole leaks and other imperfections becomes proportionally less than the laminar flow through the media. Filters therefore increase in efficiency with increasing air flow rates. When an in-place test, carried out in accordance with NRC requirements, shows an unacceptable efficiency, leakage paths can be detected by passing the aerosol through the system, and probing the downstream side of the bank of filters and mounting frame with a probe connected directly to the photometer.

Carbon filters will not be contaminated with DOP, and will be removed from the system before any testing takes place.

For small charcoal filter installations, filter bank efficiencies are determined using Freon 112, in accordance with the procedures described in DP1082 "Standardized Nondestructive Test of Carbon Beds for Reactor Confinement Applications." For large installations, the use of this procedure would necessitate the release of excessive amounts of Freon 112 within the Containment. Due to problems of possible fluoride formation, it is desirable to keep freon contamination to a minimum.

Consequently, instead of introducing freon into a fully operating ventilation system, carbon filter installations are tested a few cells at a time. The procedure is to use a small temporary portable blower and duct on the inlet side, while checking for leakage on the downstream side of the installation with a halogen leak detector. Any freon pickup which may occur in the section of the filter under test will be released following the completion of the test and will have no effect on filter performance.

The dampers and blow-in door on each air handling unit can be operated periodically to assure continued operability.

<u>System Testing</u>

Each fan cooling unit was tested after installation for proper flow and distribution through the duct distribution system. A minimum of two of the fan cooling units are used during normal operation. (Five will only be required for normal operation at design conditions, i.e., when the service water inlet temperature is above 85°F, and this condition is expected to exist only for relatively short periods, if at all.) The fan not in use can be started from the Control Room to verify readiness. The dampers and blow-in door directing flow through the carbon filter banks are tested only when the fan is not running.

After reinstallation following testing, the carbon filter units are tested in place by aerosol injection to determine integrity of the flow path.

Operational Sequence Testing

The test described in Section 6.2.5 serves to demonstrate proper transfer and sequencing of the fan motor supplies from the diesel generators in the event of loss of power. A test signal is used to demonstrate proper damper motion and fan starting prior to installation of the carbon filters. The test verifies proper functioning of the vane-switch flow indicators.

Verification of Heat Removal Capability

Since river water is circulated through the containment fan coolers and since the fans are used under both normal and accident conditions, provisions were made for verifying that the fan cooler heat removal capability does not degrade below that assumed in the containment integrity evaluation.

Instrumentation provided to verify heat removal capability is:

1) An Environmentally qualified RTD is installed on the inlet line to provide indication on the critical function monitoring system (CFMS).

2) Flow measurement of each fan cooler service water effluent is provided by an indicating flow transmitter installed in each line. The transmitter actuates a common annunciator alarm in the Control Room upon the decrease of flow in any fan cooler line.

   In addition, the flow indicator provides for manual balance of flow rates in all five fan coolers.

3) In the event of fan cooler coil service water out-leakage, the head of water will increase in a stand pipe weir which collects condensate runoff from each of the fan cooler, motor heat exchanger and demister (moisture separator).

   The increase of head is measured by a differential pressure transmitter. The current output signal is connected to an alarm unit which actuates a control room annunciator. Through the use of a weir level indicator and selector switch, the operator can determine the location of the leakage.

4) The containment building ambient temperature is controlled by manually modulating the service water flow through the fan coolers.

   The indicating range is 40° - 400°F. Average temperature indication is available at the QSPDS display and at the CR Supervisory Panel. Individual temperatures for each RTD are displayed at the CFMS. An increase in ambient temperature indicates fan cooler failure or service water discharge control valve malfunction. Either cause can be easily checked. To ensure reliability of the temperature instruments performs a channel check daily and a channel calibration every 24 months.

References

1) "Connecticut-Yankee Charcoal Filter Tests," CYAP 101, (December 1966).

2) Ackley, R.D. and R. E. Adams, "Trapping of Radioactive Methyl Iodide from Flowing Steam-Air: Westinghouse Test Series," ORNL-TM-2728, (December 1969).

3) Reactor Containment Fan Cooler System Technical Manual, Nuclear Technology Division of Westinghouse Electric Corporation, PE-1275, (May 1982).

4) Attachment I to IPN-89-046, "Proposed Change to Technical Specifications to Increase the Design Bases Ultimate Heat Sink Temperature."

Containment overpressure can be relieved as required through the pressure relief duct and exhaust fan, passing up the discharge duct, along with the exhaust air from the Primary Auxiliary Building. A narrow range pressure indicator is provided on the local fan panel to assist in operation of the building pressure relief fan. The range is –5 to +5 psig.

Components

All associated components, piping, and structures, of the Containment Penetration and Weld Channel Pressurization System are designed to Class I seismic criteria.

The piping and valves for the system are designed in accordance with the ANSI Code for Pressure Piping (Power Piping Systems), B31.1.

For a description of the instrument and control air compressors and the plant air compressor, see Section 9.6.

The three nitrogen cylinders provided meet the requirements of Section VIII (Unfired Pressure Vessels) of the ASME Boiler and Pressure Vessel Code, for 2200 psig maximum pressure, and contain a total of 22,000 scf of nitrogen.

6.6.3 Design Evaluation

The employment of this system following a Loss-of-Coolant Accident, while not considered in the analysis of the consequences of an accident, provides an additional means for ensuring that leakage is minimized, if not altogether eliminated. No detrimental effect of any other safety features system will be felt should the pressurization system fail to operate.

System Response

WCCPPS is not single failure proof, as the WCCPPS zones are not redundant. A nitrogen or air regulator failure may render a zone, or in some instances the entire system, incapable of performing its design function (i.e., pressurize the space between containment isolation valves, weld channels and containment penetrations at a pressure greater than the containment accident pressure profile for 24 hours post accident).

This can be tolerated for the following reasons. While one of the design basis functions for WCCPPS is to minimize offsite releases, WCCPPS is not needed to meet the requirements of 10 CFR 100. In addition, no other safeguards systems are dependent on the operation of WCCPPS. As such, a WCCPPS failure would not create undue risk to the health and safety of the public.

To account for active failures, two parallel WCCPPS supply valves are provided for certain containment isolation lines that are normally or intermittently open during operation. These containment isolation valves automatically close on a containment isolation signal. Opening of one of the two WCCPPS supply valves in each line is sufficient to accomplish pressurization gas injection upon closing of the containment isolation valves.

Shared Functions Evaluation

Table 6.6-4 is an evaluation of the main components discussed previously and a brief description of how each component functions during normal operation and during an accident.

6.6.4 Minimum Operating Conditions

The Technical Specifications establish limiting conditions regarding the operability of the system when the plant is above MODE 5.

6.6.5 Inspections and Tests

Inspections

The system components located outside the Containment can be visually inspected at any time. Components inside the Containment can be inspected during shutdown. All pressurized zones have provisions for either local pressure indication outside the Containment or remote low pressure alarms in the Control Room.

Testing

Since the system is in operation continuously during all reactor operations to maintain the penetrations and liner weld channels pressurized above containment design pressure, no special testing of system operation or components is necessary.

Should one zone indicate a leak during operation, the specific penetration or weld channel containing the leak can be identified by isolating the individual air supply line to each component in the zone and injecting leak test gas through a capped tube connection installed in each line.

Total leakage from penetrations and weld channels is measured by summing the recorded flows in each of the four pressurization zones. A leak would be expected to build up slowly and would therefore be noted before design leakage limits are exceeded. Therefore, remedial action can be taken before the limit is reached. For those liner welds that are no longer continuously pressurized, a leak would not be identified during plant operation. The integrity of these welds is verified by integrated leak rate testing.

In order to provide facility for testing the larger penetrations, branch pressurizing lines are provided from one of the zones to:

1) The double-gasketed space on each hatch of the Personnel Air Lock.

2) The double-gasketed space at the Equipment Hatch flange.

3) The pressurized zones in the spent fuel transfer tube.

4) The spaces between the two butterfly valves in the purge supply exhaust ducts.

6.8 HYDROGEN RECOMBINATION SYSTEM

6.8.1 Design Bases

The design bases for the hydrogen control following a postulated Loss-of-Coolant Accident are as follows:

1) The system shall prevent the hydrogen concentration in the containment volume from exceeding 3% by volume following a design basis accident.

2) The system shall be capable of performing its design function in the containment environment following a design basis accident, i.e., withstand the accident and be capable of beginning operation as required when the containment pressure is near ambient.

3) The system shall be designed to withstand the design basis earthquake and still be capable of operation.

4) The system shall be sufficiently redundant and independent to the extent that no single active or passive failure can negate the minimum requirements of operation.

5) The system shall be testable during normal operating conditions of the plant.

6.8.2 System Design and Operation

System Description

The electric hydrogen recombiner systems installed at Indian Point 3 are engineered safety features to control the hydrogen generated in the containment following a Loss-of-Coolant Accident. The redundant systems are designed to seismic Class I Standards.

Two full rated, redundant and independent systems are provided. Each recombiner is powered from a separate safety related MCC. Each is capable of maintaining the ambient H_2 concentration at or below three volume percent (v/o).

Each recombiner system consists of a control panel located in the Control Room, a power supply cabinet located in the lower electrical cable tunnel, at elevation 34 ft., and a recombiner located on the operating deck at elevation 95 ft. in the Containment. The electric hydrogen recombiners are located in the southeast and southwest quadrants of the containment approximately 90° apart in the same location as the old flame type recombiners they are replacing. There are no moving parts or controls inside the containment. Heated air within the unit causes airflow by natural convection. The recombiner is a completely passive device.

To regulate the power supply to the recombiner, the power supply cabinet contains an isolation transformer and a controller. This equipment will not be exposed to the post-LOCA environment. The controls for the power supply are located in the Control Room and are manually actuated.

Each hydrogen recombiner consists of the following components:

1) A preheater section, consisting of a shroud placed around the central heaters to take advantage of heat conduction through the central walls, for preheating incoming air.

2) An orifice plate to regulate the rate of airflow through the unit.

3) A heater section, consisting of four banks of metal-sheathed electric resistance heaters, to heat the air flowing through it to hydrogen-oxygen recombination temperatures.

4) An exhaust chamber, which mixes and dilutes the hot effluent with containment air to lower the temperature of the discharge stream.

5) An outer enclosure to protect the unit from impingement by containment spray.

The recombiner unit is manufactured of corrosion-resistant, high-temperature material. The electric hydrogen recombiner uses commercial-type electric resistance heaters sheathed with Incoloy-800 which is an excellent corrosion-resistant material for this service. The recombiner heaters operate at significantly lower power densities than similar heaters used in commercial practice.

System Operation

Each recombiner is operated from its control panel located in the Central Control Room. Emergency operating procedures direct that the hydrogen concentration in the containment be monitored (by manual sampling or with the hydrogen analyzers) following a LOCA or high containment pressure condition and that the hydrogen recombiners be actuated in time to prevent reaching a hydrogen concentration of 4.0 volume percent. System operating procedures provide instructions for the operator to manually put the recombiners in service from the control panel. The recombiner, power supply panel and control panel are shown on Figures 6.8-2 and 6.8-2A. The power panel for the recombiner contains an isolation transformer and a controller to regulate power into the recombiner. This equipment is not exposed to the post-LOCA containment environment.

To control the recombination process, the correct power input to bring the recombiner above the threshold temperature for recombination is set by adjusting a potentiometer located on the control panel. The correct power required for recombination depends upon containment atmosphere conditions and is determined when recombiner operation is required. For equipment test and periodic checkout, a temperature controller is provided on the control panel to automatically bring the recombiner to the recombination temperature.

The containment atmosphere is heated within the recombiner in a vertical duct, causing it to rise by natural convection. As it rises, replacement air is drawn through intake louvers downward through a preheater section which will temper the air and lower its relative humidity.

The preheated air then flows through an orifice plate, sized to maintain a 100 SCFM flowrate, to the heater section. The airflow is heated to a temperature above 1150°F, the reaction temperature for the hydrogen-oxygen reaction. Any free hydrogen present reacts with atmospheric oxygen to form water vapor. After passing through the heater section, the flow enters a mixing section, which is a louvered chamber where the hot gases are mixed and cooled with containment atmosphere before the gases are discharged directly into the containment. The air discharge louvers are located on three sides of the recombiner. To avoid recirculating previously processed air, no discharge louvers are located on the intake side of the recombiner. (Reference 10)

Tests have verified that the hydrogen-oxygen recombination is not a catalytic surface effect associated with the heaters but occurs due to the increased temperature of the process gases. As the phenomenon is not a catalytic effect, saturation of the unit cannot occur (References 1 and 9).

Instrumentation

The recombiners do not require any instrumentation inside the containment for proper operation after a LOCA. The recombiners are started manually after a LOCA. The sampling system is used to obtain containment atmosphere samples that indicate when the recombiners or the venting system should be actuated. Control measures can be initiated when the hydrogen concentration reaches 3.0 volume-percent.

The thermocouples and temperature transmitters located in the thermocouple splice box are used for testing and calibration only. Their failure will not affect the safety function of the recombiner.

Power Supply

Supply power for the electric recombiners is provided from safety related 480 MCC's 36C and 36B which are backed up by emergency diesel generators 31 and 32, respectively.

In order to prevent overloading of the diesel generators, the electric hydrogen recombiners will be deenergized on loss of offsite power or on a safety injection (SI) signal. Manual operator action will be required to restart the recombiners once adequate diesel generator capacity is available.

Post Accident Containment Atmosphere Sampling System

Following an accident, containment atmosphere is monitored for hydrogen concentration. A two channel redundant system is provided. Samples are taken from the plenum chambers of the containment recirculation fan units. Train A monitor takes suction from the plenum chambers of fan units 32 and 35. The train B monitor takes suction from the plenum chambers of fan units 31, 33 and 34.

Sampling should begin with the first 7 hours following diagnosis of a LOCA or MSLB inside containment. (Reference 11)

To assure that stratification effects or sample errors would not permit all or parts of the containment to hold hydrogen in excess of the lower flammable limit (4.1 v/o) when the measured concentration is 3.0 v/o, the following checks were made: It was determined that the minimum reliable air circulation capacity by three of the main recirculation fans within the containment could recirculate the entire containment air volume at an average rate of 4.8 times an hour (or 210,000 cfm capacity based upon pressure decay to ambient conditions for fan operation). But the calculated hydrogen generation rate during the first day post accident is 17,100 SCF yielding a ratio of air circulation to hydrogen generation in excess of 17,7000:1. Due to the decreased rate of hydrogen generation with time, the ratio increases to an even greater value before the hydrogen concentration in the containment reaches two volume percent. At a conservatively predicted generation rate, 60 hours are required to produce hydrogen in the amount of two percent of containment volume. During this same period, the entire atmosphere of the containment would have been recirculated, on the average, 288 times. Furthermore, the air handling system is designed to promote the interchange of air in all regions of the containment to avoid the possibility of accumulation of hydrogen in stagnant pockets or strata. For example, in the highest part of the containment dome (above the top spray ring), minimum air recirculation provides one air change approximately every 61 seconds. For these reasons it is concluded that the stratification error is negligible.

Based on the foregoing discussion, it is concluded that the three volume percent design concentration for operating the recombiner provides more than adequate margin for error associated with sampling the containment atmosphere. The calculated containment hydrogen concentration does not reach three volume percent until 10 days post accident, so it is highly unlikely that any significant concentration gradient will exist in the containment when the recombiner is started. Furthermore, since tests have been run with a full scale recombiner system at hydrogen concentrations up to and including 4.0 volume percent hydrogen, a hydrogen concentration between 2 and 3.5 volume percent at the recombiner suction would have no adverse effect on the recombiner operation.

6.8.3 Design Evaluation

The analysis of post-LOCA hydrogen production and accumulation in the containment is presented in Section 14.3.7. To determine the effectiveness of the recombiner, it is assumed that it will be activated before the containment hydrogen concentraiton reaches the design limit of 3 volume-percent. Starting the recombiner at below 3% provides substantial margin in time to reach the lower flammability concentration of 4.1%. The capacity of the recombiner, working in a 3% hydrogen environment, is at least 3 SCFM of hydrogen gas.

The results of the Regulatory Guide 1.7 analysis indicate that 3% hydrogen occurs at approximately 10 days (Figure 14.3-117) and the corresponding aggregate hydrogen production rate is approximately 1.6 SCFM (Figure 14.3-113). This production rate is well within the capacity of the recombiner. Further, because the hydrogen production rate decreases with time, the recombiner can easily accommodate hydrogen concentrations greater than 3%. Thus, starting a recombiner before the containment hydrogen concentration reaches 3% will ensure that the concentration remains well below the lower flammability limit.

Hydrogen stratification in the containment post-LOCA is minimized by the operation of the containment fan coolers. The containment coolers circulate air within the containment volume (Section 6.8.2). A containment sampling line is located near the inlet of each fan cooler. Assuming that 3 of 5 fan coolers are operating and that the flow rate per unit for the first day is 34,000 cfm (design flow rate during accident conditions, Table 5.3-1) and 70,000 cfm thereafter (design flow rate during normal operation, Table 5.3-1) results in an average air flow of approximately 36 containment volumes per day (based on 10 days of operation) for each fan.

The recombiners are located in an open area of the containment on the 95' elevation operating deck.

The calculated average hydrogen concentration in containment reaches three volume percent in approximately 10 days (Safety Guide 7 basis, Figure 14.3-117). Based on this relatively small hydrogen production rate (average less than 5 SCFM, approximately 1.6 SCFM at 10 days, Figure 14.3-113) and the large air mixing rate described above, the bulk of the containment volume is expected to be well mixed, and no significant hydrogen concentration gradients are expected at either the hydrogen sampling points or at the recombiner locations.

Personnel Doses

The control panel for the hydrogen recombiner is located in the Control Room. The control room is designed to provide radiation protection for the operator following a design basis event. Doses to the control room operators following a large-break LOCA are evaluated in Section 14.3.5. The calculated doses are well within the limits specified in General Design Criterion 19, i.e., 30 rem thyroid and 5 rem whole body.

6.8.4 Tests and Inspections

The electric hydrogen recombiners underwent extensive testing in the Westinghouse development program. These tests encompassed the initial analytical studies, laboratory proof-of-principle tests, and full-scale prototype testing. The full scale prototype tests include the effect of:

- Varying hydrogen concentrations
- Alkaline spray atmosphere
- Steam effects
- Convection currents
- Seismic effects

A detailed discussion of these tests is provided in references 1 through 9.

Operational tests and inspections are performed in accordance with the requirements of the Technical Specifications to verify the operation of the control system and the ability of the heaters to achieve the required temperature. In addition, a channel calibration of all recombiner instrumentation and control circuits is performed every 24 months.

References

1) Wilson, J. F., "Electric Hydrogen Recombiner for Water Reactor Containments," WCAP-7709-L (Proprietary), July 1971, and WCAP-7820 (Nonproprietary), December 1971.

2) Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments – Final Development Report," WCAP-7709-L Supplement 1 (Proprietary), and WCAP-7820, Supplement 1 (Nonproprietary), April 1972.

3) Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments – Equipment Qualification Report," WCAP-7709-L Supplement 2 (Proprietary), and WCAP-7820, Supplement 2 (Nonproprietary), September 1973.

4) Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments – Long Term Tests," WCAP-7709-L Supplement 3 (Proprietary), and WCAP-7820, Supplement 3 (Nonproprietary), January 1974.

5) Wilson, J. F., "Electric Hydrogen Recombiner for PWR Containments," WCAP-7709-L Supplement 4 (Proprietary), and WCAP-7820, Supplement 4 (Nonproprietary), April 1974.

6) Wilson, J. F., "Electric Hydrogen Recombiner Special Tests," WCAP-7709-L Supplement 5 (Proprietary), and WCAP-7820, Supplement 5 (Nonproprietary), December 1975.

7) Wilson, J. F., "Electric Hydrogen Recombiner IEEE 323-1974 Qualification," WCAP-7709-L Supplement 6 (Proprietary), and WCAP-7820, Supplement 6 (Nonproprietary), October 1976.

8) Wilson, J. F., "Electric Hydrogen Recombiner for LWR Containments Supplemental Test Number 2," WCAP-7709-L Supplement 7 (Proprietary), and WCAP-7820, Supplement 7 (Nonproprietary), August 1977.

9) Wilson, J. F., "qualification Testing for Model B Electric Hydrogen Recombiner," WCAP-9346 (Proprietary) and WCAP-9347 (Nonproprietary), July 1978.

10) Electric Hydrogen Recombiner Model B Technical Manual, NYPA File 439-100058911.

11) NSE 97-3-289 HR

Ch. 7

The QSPDS design and display is based on NRC Regulatory Guide 1.97 criteria. The CFMS provides for historical data storage and retrieval capability (HDSR). The HDSR system will record, store, recall and display historical information either as graphs and trends or printed logs.

The CFMS/QSPDS receive signals from various plant equipment. The CFMS receives signals from safety related and non-safety related sources, and adequate electrical separation is maintained by use of fiber optic links.

In order to comply with the requirements of Regulatory Guide 1.97, additions to the original plant design parameters were made. Transmitters monitoring many process variables were installed and the CFMS is utilized to alarm and display these parameters. In some cases local indicators are also provided to facilitate local operation needs. Besides additions, replacement of existing components were made to upgrade them to meet the requirements.

7.5.3 System Evaluation

Redundant instrumentation has been provided for all inputs to the protective system and vital control circuits.

Where wide process variable ranges and precise control are required, both wide range and narrow range instrumentation is provided.

All electrical and electronic instrumentation required for safe and reliable operation is supplied from four redundant instrumentation buses.

7.5.4 Instrument Required

Table 7.5-1 identifies the instruments used to demonstrate compliance with NRC Regulatory Guide 1.97. Exemptions to compliance are noted in the table.

The Technical Specifications establish required actions and completion times for Regulatory Guide 1.97 Type A and Category 1 instrument channels.

In addition, inoperability of the following associated recorders is limited to 14 days: Containment Pressure, Containment Water Level, Recirculation Sump Water Level, Containment Hydrogen Monitor, Steam Generator Water level (Wide Range), RCS Pressure (Wide Range), Cold Leg Temperature (Wide Range), Hot Leg Temperature (Wide Range), Pressurizer Water Level, RCS Subcooling Monitor.

Surveillance requirements for Regulatory Guide 1.97 Type A and Category 1 instruments are established in the Technical Specifications. In addition, a Channel Operational Test is required, as follows, for alarms that are associated with Type A and Category 1 instruments, but which have no Regulatory Guide function:

- Main Steam Line Radiation (R62), Quarterly
- Gross Failed Fuel Detector (R63), Quarterly
- Containment Hydrogen Monitor, Monthly

Control and Readout Description

The control and readout system provides means for inserting the miniature neutron detectors into the reactor core and withdrawing the detectors at a selected speed while plotting a level of induced radioactivity versus detector position. The control system consists of two sections, one physically mounted with the drive units, and the other contained in the control room. Limit switches in each path provide feedback of path selection operation. Each gear box drives an encoder for position feedback. One 5-path group selector is provided for each drive unit to route the detector into one of the flux thimble groups. A 10-path rotary transfer assembly is a transfer device that is used to route a detector into any one of up to ten selectable paths. Manually operated isolation valves allow free passage of the detector and drive wire when open, and prevents steam leakage from the core in case of a thimble rupture, when closed. A common path is provided to permit cross calibration of the detectors.

The control room contains the necessary equipment for control, position indication, and flux recording. Panels are provided to indicate the core position of the detectors, and for plotting the flux level versus the detector position. Additional panels are provided for such features as drive motor controls, core path selector switches, plotting and gain controls. A "flux-mapping" consists, briefly, of selecting (by panel switches) flux thimbles in given fuel assemblies at various core quadrant locations. The detectors are driven or inserted to the top of the core and stopped automatically. A x-y plot (position vs. flux level) is initiated with the slow withdrawal of the detectors through the core from top to a point below the bottom. In a similar manner other core locations are selected and plotted.

The system that will be used to monitor the distribution of power in the X-Y plane is described in WCAP-7669, "Topical Report – Nuclear Instrumentation System."

Operational limits due to a quadrant power tilt are given in the Technical Specifications.

The calibration of the Nuclear Instrumentation System by the movable incore detector system is made in accordance with the Technical Specifications. As noted in the Technical Specifications, the movable incore detector system shall be used to confirm power distribution.

After the excore system is calibrated initially, recalibration is performed periodically to compensate for changes in the core, due for example to fuel depletion, and for changes in the detectors.

If the recalibration is not performed, the mandated power reduction assures safe operation of the reactor as it will compensate for an error of 10% in the excore protection system. Experience at Baznau No. 1 and R. E. Ginna plants has shown that drift due to changes in the core or instrument channels is very slight. Thus the 10% reduction is considered to be very conservative.

The reactor trip functions (Section 7.2) provide core protection at the safety limits prescribed in the Technical Specifications. Those trip functions derived from the Nuclear Instrumentation System are described in WCAP-7669.

Each detector provides axial flux distribution data along the center of a fuel assembly. Various radial positions of detectors are then compared to obtain a flux map for a region of the core

7.6.3 System Evaluation

The thimbles are distributed throughout the core as shown in Figure 7.6-1. The positions have been chosen to provide symmetry checks and sufficient coverage, taking symmetry into account, to construct a full core three-dimensional power shape. With this number and location of thimbles the measurement accuracy for the peak to average rod in an x-y plane is 3.65% and for the peak to average pellet, including axial peaking, is 4.58%. These accuracies include the flux thimble to hot rod calculational uncertainty and instrumentation repeatability. They represent a 95% confidence level in a probability of fewer than 5% of cases lying above this error allowance. This confidence level and accuracy is consistent with the interpretation of DNB criteria.

The derivation and justification of these uncertainties is given in WCAP-7308-L, "Evaluation of Nuclear Hot Channel Factor Uncertainties."

7.6.4 System Operation

A. A minimum of 2 thimbles per quardrant and sufficient movable incore detectors shall be operable during recalibration of the excore axial offset detection system.

B. During the incore / excore calibration procedure, full core flux maps will be made only when at least 38 of the movable detector guide thimbles are operable.

7.7 OPERATING CONTROL STATIONS

7.7.1 Station Layout

The principal criteria of control station design and layout is that all controls, instrumentation displays and alarms required for the safe operation and shutdown of the plant are readily available to the operators in the Control Room.

During other than normal operating conditions, other operators will be available to assist the operators in the Control Room. Figure 7.7-1 and 7.7-2 show the Control Room layout and sections for the unit. The control board is divided into relative areas to show the location of control components and information display pertaining to various subsystems.

7.7.2 Information Display and Recording

Alarms and annunciators in the Control Room provide the warning to the operators of abnormal plant conditions which might lead to damage of components, fuel or other unsafe conditions. Other displays and recorders are provided for indication of routing plant operating conditions and for the maintenance of records.

Consideration is given to the fact that certain systems normally require more attention from the operator. The control system is therefore centrally located on the three section board.

On the left section of the control board, individual indicators present a direct, continuous readout of every control rod position. Fault detectors in the rod drive control system are used to alert the operator should an abnormal condition exist for any individual or group of control rods. Displayed in this same area are limit lights for each control rod group and all nuclear instrumentation information required to start up and operate the reactor. Control rods are manipulated from the left section.

Subsequent to periods of rod motion, when thermal equilibrium is being established in the rod position indicator coil stacks, temporary drifting of the indicators can be expected. During such time if indicated RCCA position differs from bank demand more than allowed by the Technical Specifications, the rod is treated as potentially misaligned under Technical Specification 3.1.4. Rod position is confirmed via a digital voltage meter applied to the rod position control racks. In addition, the operators will continue to monitor the affected rod position indicators on the main control board (and on the plant computer, if available and in agreement with the digital voltage meter reading) to check for increased deviation.

Variables associated with operation of the secondary side of the station are displayed and controlled from the control board. These variables include steam pressure and temperature, feedwater flow, electrical load, and other signals involved in the plant control system. The control board also contains provisions for indication and control of the reactor coolant system. Redundant indication is incorporated in the system design since pressure and temperature variables of the Reactor Coolant System are used to initiate safety features. Control and display equipment for station auxiliary systems is also located here.

The Engineered Safety Features Systems are controlled and monitored form a vertical panel to the left of the control board. Valve position indicating lights are provided as a means of verifying the proper operation of the control and isolation valves following initiation of the engineered safety features. Control switches located on this panel allow manual operation or test of individual units.

Also located on this section are the control switches, indicating lights, and meters for fans and pumps required for emergency conditions. Also mounted on this section are auxiliary electrical system controls required for manual switching between the various power sources described in Section 8.2.2.

Controls and indications for all ventilation systems, the containment isolation valves, and the Isolation Valves Seal Water System are located on a vertical panel. Radiation monitoring information is indicated immediately behind and to the left of the main control board.

Audible Reactor Building alarms are initiated from the radiation monitoring system and from the source range nuclear instrumentation. Audible alarms will be sounded in appropriate areas throughout the station if high radiation conditions are present.

7.7.3 Emergency Shutdown Control

The Control Room, its equipment and furnishings were designed so that the likelihood of fire or other conditions which could render the Control Room inaccessible even for a short time is extremely small. For details on the fire protection features, refer to Section 9.6.2.

A criterion of the station design and layout was that all controls, instrumentation displays and alarms required for the safe operation and shutdown of the plant are readily available to the operators in the Control Room.

It was design policy that the functional capacity of the Control Room should be maintained at all times inclusive of accident conditions, such as a Maximum Credible Accident or a fire; the following features were incorporated in the design to ensure that this criterion was met.

Structural and finish materials for the Control Room and the cable spreading room below were selected on the basis of fire resistant characteristics. Structural floors are concrete reinforced. Interior partitions are metal paneling joints. The Control Room ceiling covering is fire retardant egg crate diffusers. Door frames and doors are metallic. Wooden trim is not used.

The Control Room is equipped with portable fire extinguishers sized and located in accordance with National Fire Code and National Fire Protection Association specifications. Extinguishers carry the Underwriter's Laboratory label of approval and are electrical shock resistant.

Fire protection features of the cable spreading room and safe shutdown capability in the event of a fire in the cable spreading room are discussed in Section 9.6.2.

7.9 SURVEILLANCE REQUIREMENTS

The requirements for periodic testing of instruments are listed in the Technical Specifications, Technical Requirements Manual, the FSAR, and the ODCM. The type of test action (i.e., channel calibration, channel operational test, etc.) to be taken and the minimum testing frequency (i.e., 31 days, 92 days, 24 months, etc.) for the indicated instruments are provided within the above-mentioned documents.

As indicated, the instrumentation channels which are covered include, for example, nuclear, reactor coolant temperature and flow, pressurizer pressure and level, and auxiliary process channels; or components necessary to assure that facility operation is maintained within the safe limits.

Ch.

8

The recirculation phase is manually initiated by control switches on the supervisory panel in the main control room. As the sequence switches are operated, the bus loads are modified to give those shown in Reference 1 for the respective Design Basis Accidents.

Emergency Diesel Generator Loading

The following "minimum safeguards" equipment is required and assumed to be operating for a design basis event at Indian Point Unit 3:

2 of 3 Safety Injection (SI) Pumps
1 of 2 Residual Heat Removal (RHR) Pumps
1 of 2 Motor Driven Auxiliary Feedwater (AFW) Pumps
1 of 2 Recirculation Pumps
3 of 5 Containment Recirculation (CR) Fans
1 of 2 Containment Spray (CS) Pumps
1 of 3 Nonessential Service Water (NE SW) Pumps
2 of 3 Essential Service Water (ESW) Pumps
1 of 3 Component Cooling Water (CCW) Pumps

Due to interactions between systems, minimum requirements for safety vary with the loss of any one diesel generator. See Chapter 14.3 for details.

This configuration is based on the assumptions of a single active failure of an emergency diesel generator and that 1 CCW and 1 NE SW pump may be out of service at the time of the accident. In addition to the required equipment listed above, the operator may manually load other equipment during the recovery process as instructed by the Emergency Operating Procedures (EOPs) or System Operating Procedures (SOPs).

The maximum steady state power requirements for equipment that is either automatically or manually loaded in the emergency diesel generators following a loss of offsite power and SI actuation have been conservatively calculated in Reference 1. The diesel generator loading in each of the following design base accidents; large break loss of coolant, small break loss of coolant, main steam line break, and steam generator tube rupture are evaluated in Reference 1 for the actual sequence of loading that the control room operators would initiate as they respond to a DBA. In the initial stage for the worst case accident, the peak load is less than 1950 kW. As the plant approaches steady state (accident stabilized) conditions the EDG loading is less than the unit 1750 kW continuous rating. The maximum steady state power requirements for equipment loading in the emergency diesel generators following a reactor trip without engineered safeguard actuation (SI) with loss of offsite power have been conservatively calculated in Reference 2. Similar to the SI accident scenarios, at the initial stage of the accident the peak load is less than 1950 kW. At steady state, the diesel load is less than 1750 kW. Equipment loading range on the EDG's for both the SI and Non-SI accidents is summarized in Table 8.2-1A.

The worst case transient loading histories were computed assuming the possibilities of a diesel failure combined with equipment out of service.

Design basis events which do not actuate the safety injection system will result in lower emergency diesel loading than those that do.

Testing

To verify that the emergency power system will respond within the required time limit and when required, the following tests shall be performed periodically.

a) Manually initiated demonstration of the ability of the diesel generators to start and deliver power up to nameplate rating when operating in parallel with other power sources. Normal plant operation will not be affected. The duration of the test is at least one hour to at least 50% of continuous rating.

b) Demonstration of the readiness of the system and the control systems of vital equipment to automatically start or restore to operation particular vital equipment by simulating a loss of all normal AC station service power supplies. This test is conducted as required by the Technical Specifications.

The starting of the diesel generator sets can be tested from the Diesel Generator Building. The ability of the units to start within the prescribed time and to carry intended loads are checked periodically. (See Section 8.5).

In addition, each diesel generator shall be inspected and maintained following the manufacturer's recommendations for this class of standby service.

Battery No. 31	floating charge equalizing charge	>23 hours >3 hours
Battery No. 32	floating charge equalizing charge	>30 hours >3 hours
Battery No. 33	floating charge equalizing charge	>77 hours >7.7 hours
Battery No. 34	floating charge equalizing charge	>11.5 hours >3.8 hours
Battery No. 36	floating charge equalizing charge	>17 hours >7.8 hours

The ventilation system for Battery Rooms No. 31, 32, 34 and 36 operate continuously, to preclude any hydrogen build-up. (Station Battery No. 33 is located in Diesel Generator Room No. 31 and does not require forced ventilation.) Loss of the battery room ventilation is annunciated in the Control Room; loss of diesel operating room ventilation is detected by supervisory personnel observations and/or normal operating maintenance procedures.

Normally the batteries are on continual floating charge. They are placed on a 24 hour equalizing charge every quarter or after an emergency battery discharge. (Manual actuation is required at the battery chargers to place the batteries on equalizing charge.)

There is one (1) annunciator window labeled "Battery Charger Trouble." This alarm is set off on the following signals from battery chargers as indicated:

SIGNAL	CHARGERS
1) Low DC Voltage	31, 32, 33, 34, 35, & 36
2) Ground Detection	31, 32, 33, 34, 35, & 36
3) AC Power Failure	31, 32, 33, 34, 35
4) High-Low AC Voltage	36
5) Over Temp	31, 32, 35 & 36
6) High DC Voltage	36
7) High DC Voltage Shutdown	31, 32, & 35
8) Battery Discharge	31, 32, 35 & 36
9) Charger Failure	36

Each individual signal can be isolated on each individual charger listed. Indication is provided at each charger when at any signal is isolated on that charger.

Reliability Assurance

The electrical system equipment is arranged so that no single contingency can inactivate enough safeguards equipment to jeopardize the plant safety. The 480-volt safeguards equipment is arranged on 4 buses. The 6900-volt equipment is supplied from 7 buses.

The plant auxiliary equipment is arranged electrically so that multiple items receive their power from the two different sources. The charging pumps are supplied from the 480 volt buses No. 3A, 5A and 6A. The nine service water pumps and the five containment fans are divided among five of the 480-volt buses. Valves are supplied from motor control centers, No. 36A and 36B, which are supplied from buses No. 5A and 6A.

The outside source of power is adequate to run all normal operating equipment. The 138 kV – 6.9 kV station transformer can supply all the auxiliary loads.

The bus arrangements specified for operation ensure that power is available to an adequate number of safeguards auxiliaries.

Minimum engineered safeguards can be carried by any two diesel generators. These safeguards can adequately cool the core and maintain containment pressure within the design value for the Design Basis Accident.

One battery charger is available to each battery so that the four batteries will always be at full charge in anticipation of loss-of-AC power incident. This ensures that adequate DC power will be available for starting emergency generators and other emergency uses.

8.2.4 Engineered Safeguards Components

The initiation, control and sequencing design of engineered safeguards components, Auxiliary Feedwater System, and Component Cooling Water System is as shown on the schematics listed on Table 8.2-3.

8.3 MINIMUM OPERATING CONDITIONS

The minimum operating conditions for electrical systems are given in Sections 3.8.1 and 3.8.10 of the Technical Specifications.

8.5 TESTS AND INSPECTIONS

The tests discussed in this Section are designed to demonstrate that the Diesel Generators will provide power for operation of equipment. They also assure that the emergency generator system controls and the control systems for safeguards equipment will function automatically in the event of a loss of all normal 480 volt AC station service power.

The testing frequency dictated by the Technical Specifications provides for testing often enough to identify and correct deficiencies to systems under test before they can result in a system failure. The fuel supply and starting circuits and controls are continuously monitored and any faults are indicated by alarms. An abnormal condition in these systems would be signaled without having to place the Diesel Generators themselves on test.

To verify that the emergency power system does respond properly and within the required time limit when required, the following tests are performed periodically:

a) Manually initiated demonstration of the ability of the Diesel Generators to start, and deliver power up to name plate rating, when operating in parallel with other power sources. Normal plant operation will not be affected. The duration of the test shall be at least one hour to at least 50% of continuous rating.

b) Demonstration of the readiness of the system and control systems of vital equipment to automatically start or restore to operation particular vital equipment by initiating an actual loss of all normal AC station service power supplies. This test is conducted as dictated by the Technical Specifications.

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

To verify that the 480 V safeguards bus undervoltage alarms operate properly they shall be tested monthly and calibrated every 24 months.

In addition, each diesel generator shall be inspected and maintained following the manufacturer's recommendations for this class of standby service.

Ch.

9

A non-Chromate based corrosion inhibitor is added to the Component Cooling water system. This inhibitor contains a compound for corrosion control on non-ferrous materials.

The fluorides are kept below 0.15 ppm each, chlorides are kept below 150 ppm, and the makeup is of reactor coolant water quality. Experience at other operating plants has shown that sodium molybdate corrosion inhibitors as a whole are effective in controlling corrosion of carbon, alloy and stainless steel.

Assurance of proper component cooling water chemistry is provided through periodic sampling. The Component Cooling Water is sampled and analyzed at least monthly for gross activity, corrosion inhibitor and pH. The maximum time between analyses is 45 days.

Residual Heat Removal Loop

The residual heat removal loop, as shown in Figures 9.2-1 and 9.3-2A consists of heat exchangers, pumps, piping and the necessary valves and instrumentation. During plant shutdown, coolant flows from the Reactor Coolant System to the residual heat removal pumps, through the tube side of the residual heat exchangers and back to the Reactor Coolant System. The inlet line to the residual heat removal loop starts at the hot leg of one reactor coolant loop and the return line connects to the Safety Injection System piping. The residual heat exchangers are also used to cool the water during the latter phase of Safety Injection System operation. These duties are defined in Chapter 6. The heat loads are transferred by the residual heat exchangers to the component cooling water.

During plant shutdown, the cooldown rate of the reactor coolant and the component cooling water heat exchanger outlet temperature are controlled by regulating the flow through the tube side of the residual heat exchangers. Two remotely operated control valves, downstream of the residual heat exchangers, are used to control flow. Manual throttle valves are used to control component cooling water flow to the residual heat removal heat exchangers and service water flow to the component cooling water heat exchangers.

Double, remotely operated valving is provided to isolate the residual heat removal loop from the Reactor Coolant System. Whenever the reactor coolant system pressure exceeds the design pressure of the residual heat removal loop, separate reactor coolant system pressure channel interlocks will automatically close these valves. Inn addition, the interlocks also prevent the valves from opening until a predetermined set point is reached. Two remotely operated valves and one check valve isolate each line to the Reactor Coolant System cold legs from the residual heat removal loop.

Recirculation lines that branch off from the RHR pump discharge piping upstream of the discharge check valves have been installed to ensure a minimum pump recirculation flow of 300 gpm. This recirculation line configuration also eliminates the possibility of "dead heading" an RHR pump during dual pump operation by effectively separating the two pump discharge lines.

Spent Fuel Pit Cooling Loop

The spent fuel pit cooling loop removes residual heat from fuel stored in the spent fuel pit. The loop is normally required to handle the heat load from 76 freshly discharged fuel assemblies from the reactor, but it can safely accommodate the heat load from all of the 1345 assemblies for which there is storage space available.

The spent fuel is placed in the pit during refuelings for long-term storage. The spent fuel pit capacity allows for storage of 6 cores while retaining enough capacity for a full core unload.

The spent fuel pit is located outside the Reactor Containment and is not affected by any Loss-of-Coolant Accident in the Containment. During refueling, the water in the pit is connected to that in the refueling canal by the fuel transfer tube. Only a very small amount of water interchange occurs as fuel assemblies are transferred.

The spent fuel pit cooling loop consists of pumps (main and standby), heat exchanger, filters, demineralizer, piping and associated valves and instrumentation. The operating pump draws water from the pit, circulates it through the heat exchanger and returns it to the pit. Component cooling water cools the heat exchanger. A second pumping system is used to circulate refueling water through the demineralizer and filter for purification. This is permitted under administrative controls (i.e., a trained and dedicated person to contact with the control room). Redundancy of this equipment is not required because of the large heat capacity of the pit and the slow heat up rate as shown in Table 9.3-3. However, connections are provided for an additional future heat exchanger. In the event of a failure of the spent fuel pump, the standby pump can be put into operation immediately from a local startup push button station.

In addition, reactor cavity filter tie-ins have been added to the spent fuel pit cooling loop to assist in purifying refueling water as it is drained from the reactor cavity to minimize the concentration of particulates in the refueling water storage tank.

The clarity and purity of the spent fuel pit water are maintained by passing approximately 5 percent of the loop flow through a filter and demineralizer. The spent fuel pit pump suction line, which is used to drain water from the pit, penetrates the spent fuel pit wall above the fuel assemblies. The penetration location prevents loss of water as a result of a possible suction line rupture.

Component Cooling Loop Components

Component Cooling Heat Exchangers

The two component cooling heat exchangers are of the shell and straight tube type. Service water circulates through the tubes while component cooling water circulates through the shell side. The outlet water temperature of the component cooling heat exchangers is controlled manually by throttling the service water throttle valves. Design parameters are presented in Table 9.3-1.

Component Cooling Pumps

The three component cooling pumps which circulate component cooling water through the component cooling loop are horizontal, centrifugal units. The pump casings were made from cast iron (ASTM 48) based on the corrosion-erosion resistance and the ability to obtain sound castings. The material thickness is indicated by high quality casting practice and ability to withstand mechanical damage and, as such, was substantially overdesigned from a stress level standpoint. Design parameters are presented in Table 9.3-1.

Auxiliary Component Cooling Pumps

A minimum of two of the four auxiliary component cooling pumps are automatically started during the injection phase to protect the internal recirculation pump motors from the containment atmosphere. A booster pump is also connected to the motor shaft of each safety injection pump to cool the safety injection pump bearings. The auxiliary component cooling water pumps are located outside the Containment and are seismic Class I. For further discussion of the auxiliary component cooling pump refer to Section 6.2.2. Design parameters are presented in Table 9.3-1.

Component Cooling surge Tanks

The component cooling surge tanks, which accommodate changes in component cooling water volume, were constructed of carbon steel. Design parameters are presented in Table 9.3-1. In addition to piping connections, the tanks have a flanged opening at the top for the addition of the chemical corrosion inhibitor to the component cooling loop.

The internals of the relief valves have been removed to provide a direct path to the Waste Holdup Tanks to prevent a potential overpressurization of the component cooling system.

Component Cooling Valves

The valves used in the component cooling loop are standard commercial valves constructed of carbon steel with bronze or stainless steel trim. Since the component cooling water is not normally radioactive, special features to prevent leakage to the atmosphere are not provided.

Self-actuated spring loaded relief valves are provided for lines and components, that could be pressurized to their design pressure by improper operation or malfunction.

Component Cooling Piping

All component cooling loop piping is carbon steel with welded joints and connections except at components which might need to be removed for maintenance.

Residual Heat Removal Loop Components

Residual Heat Removal Heat Exchangers

The two residual heat removal heat exchangers located within the Containment are of the shell and U-tube type with the tubes welded to the tube sheet. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell side. The tubes and other surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

Residual Heat Removal Pumps

The two residual heat removal pumps are vertical, centrifugal units with special seals to prevent reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

The residual heat removal pump seal heat exchangers and stuffing boxes are cooled from the Component Cooling Water System. A backup cooling water supply is provided from the city water system in the event the component cooling water loop is out of service. The residual heat removal pumps can be operated for an unlimited length of time, providing the supply of city water is uninterrupted. The residual heat removal pumps can operate without cooling for up to 24 hours in the event of an emergency.

Residual Heat Removal Valves

The valves used in the residual heat removal loop are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote and manual control of the residual heat exchanger tube side flow. Check valves prevent reverse flow through the residual heat removal pumps.

Two remotely operated series stop valves at the inlet with independent pressure interlocks isolate the residual heat removal loop from the Reactor Coolant System. The residual heat removal loop is isolated from the Reactor Coolant System by one check valve and two remotely operated stop valves on the outlet line. Overpressure in the residual heat removal loop is relieved through relief valves to the pressurizer relief tank. In addition, Technical Specification Section 3.4.12 restricts operation of the SI pumps when the RECS average cold leg temperature is below the OPS enable temperature. These restrictions help to preclude RHR overpressurization.

Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Waste Disposal System.

Manually operated valves have backseats to facilitate repacking. Valve backseats are capable of limiting the stem leakage when used.

Residual Heat Removal Piping

All residual heat removal piping is austenitic stainless steel. The piping is welded with flanged connections at the pumps.

Spent Fuel Pit Loop Components

Spent Fuel Pit Heat Exchanger

The spent fuel pit heat exchanger is of the shell and U-tube type with the tubes welded to the tube sheet. Component cooling water circulates through the shell, and spent fuel pit water circulates through the tubes. The tubes are austenitic stainless steel and the shell is carbon steel.

Flow indication is provided on the component cooling return lines from the safety injection and residual heat removal pumps. Each of the component cooling supply lines to the residual heat exchangers contains a check valve; each return line has a remotely operated valve (normally closed) outside the containment wall. If one of the valves fails to open at initiation of long-term recirculation, the valve which does open supplies a heat exchanger with sufficient cooling to remove the heat load.

The equipment vent and drain lines outside the Containment have manual valves which are normally closed unless the equipment is being vented or drained for maintenance or repair operations.

A failure of pumps, heat exchangers, and valves for the Component Cooling Water System is presented in Table 9.3-5.

<u>Residual Heat Removal Loop</u>

The residual heat removal loop is connected to one reactor hot leg on the suction side and to the reaction cold legs on the discharge side. On the suction side, the connection is through two electric, motor-operated gate valves in series. Each valve is independently interlocked with reactor coolant system pressure. On the discharge side, the connection is through two electric, motor-operated gate valves and one check valve for each reactor cold leg. The motor-operated valves are open whenever the reactor is in MODES 1, 2, or 3, in accordance with Technical Specification requirements.

<u>Spent Fuel Pit Cooling Loop</u>

The most serious failure of this loop is complete loss of water in the storage pool. To protect against this possibility, the spent fuel storage pool cooling connections enter near the water level so that the pool cannot be either gravity-drained or inadvertently drained. For this same reason, care was exercised in the design and installation of the fuel transfer tube. The water in the spent fuel pit below the cooling loop connections could be removed by using a portable pump.

9.3.4 Minimum Operating Conditions

Minimum operating conditions for the Auxiliary Coolant System are given in the Technical Specifications.

9.3.5 Tests and Inspections

The residual heat removal pumps flow instrument channels are calibrated during each refueling operation.

The check valves on the lines from the residual heat removal loop to the cold legs of the Reactor Coolant System are leak tested every time the plant is shutdown and the reactor coolant system has been depressurized to 700 psig or less. This test is also performed following valve maintenance, repair or other work which could unseat these check valves.

The portion of the Residual Heat Removal System outside containment shall be tested for leakage at least every 24 months as follows:

1. The portion of the Residual Heat Removal System that is outside the containment shall be tested either by use in normal operation of hydrostatically tested at 350 psig.

2. The piping between the residual heart removal pumps suction and the containment isolation valves in the residual heat removal pump suction line from the containment sump shall be hydrostatically tested at no less than 100 psig.

3. Visual inspection shall be made for excessive leakage during these tests from components of the system. Any significant leakage shall be measured by collection and weighing or by another equivalent method.

4. The maximum allowable leakage from the Residual Heat Removal System components and Safety Injection System components, located outside of the containment and used during the recirculation phase of design basis accident, shall not exceed two gallons per hour.

5. Repairs of isolation shall be made as required to maintain leakage within the acceptance criterion.

Delay Coil

The high pressure reactor coolant sample line is designed to be of sufficient length to provide at least a 40 seconds sample transit time within the Containment and an additional 20 seconds transit time from the Reactor Containment to the sampling hood. This allows for decay of short lived isotopes to a level that permits normal access to the sampling room.

Sample Sink

The sample sink is located in a hooded enclosure that is equipped with an exhaust ventilator. The work area around the sink and the enclosure is large enough for sample collection and for storage of radiation monitoring equipment. The sink perimeter has a raised edge to contain any spilled liquid. The enclosure is penetrated by sample lines from the reactor plant, a demineralized water line, and steam system lines, all of which discharge into the sink. The sink and work areas are stainless steel.

Piping and Fittings

All liquid and gas ample lines are austenitic stainless steel tubing and are designed for high pressure service. Compression fittings and socket welded joints are used throughout the Sampling System. Lines are so located as to protect them from accidental damage during routine operation and maintenance.

Valves

Remotely operated stop valves are used to isolate all sample points and to route sample fluid flow inside the reactor containment. Manual stop valves are provided for component isolation and flow path control at all normally accessible Sampling System locations. Manual throttle valves are provided to adjust the samples flow rate as indicated on Figure 9.4-1.

Appropriate valves and administrative controls prevent gross reverse flow of gas from the volume control tank into the sample sink.

All valves in the system are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Isolation valves are provided outside the Reactor Containment; they trip closed upon a containment isolation signal.

9.4.1.3 System Evaluation

Availability and Reliability

Neither automatic nor operator action is required for the Sampling System during an emergency or to prevent an emergency condition. The system is therefore designed in accordance with standard practices of the chemical processing industry.

Leakage Provisions

Leakage of radioactive reactor coolant from this system within the Containment is evaporated to the containment atmosphere and removed by the cooling coils of the Recirculation Air Heating and Cooling System. Leakage of radioactive material from the most likely places outside the Containment is collected by placing the entire sampling station under a hood provided with an offgas vent to the building exhaust. Liquid leakage from the valves in the hood is drained to the chemical drain tank.

Incident Control

The system operates under administrative manual control.

Malfunction Analysis

To evaluate system safety, the failures or malfunctions are assumed concurrent with a Loss-of-Coolant Accident, and the consequences analyzed. The results are presented in Table 9.4-3. From this evaluation it is concluded that proper consideration has been given to station safety in the design of the system.

9.4.1.4 Minimum Operating Conditions

Minimum operating conditions are specified in the Technical Specifications, Technical Requirements Manual, FSAR and ODCM.

9.4.1.5 Tests and Inspections

Examples for frequency of sample analyses are as follows:

a) Reactor coolant – radiochemical analysis – every seven days.

b) Reactor coolant – boron concentration – 2 days per week, Maximum 5 days between analyses.

c) Refueling water – storage tank water – boron concentration – monthly.

9.4.2 Post-Accident Sampling System

9.4.2.1 Post-Accident Reactor Coolant Sampling System

Under post-accident conditions, the radioactivity of the primary coolant may be increased by several orders of magnitude. Access into the primary sample room is prohibited by extremely high exposure rates. The post-accident reactor coolant sampling system (shown in Figure 9.4-2) provides a safe and accurate method of obtaining a pressurized coolant sample and a means for analyzing the sample of dissolved gases, hydrogen, isotopic content, chloride, and boron. Samples of the recirculation pump discharge and the residual heat removal system can also be taken.

To obtain a primary coolant sample in a post-accident condition, temporary diversion of a representative sample stream of primary coolant is made into a shielded compartment outside the sample room. The primary coolant is diverted downstream of the sample coolers through quick disconnect couplings to a sample cask in the lower portion of the shielded compartment. The shielded compartment consists of three connected compartments with hinged doors; walls and doors are steel-encased lead, with a minimum thickness of 1-1/2 inch. Channels of poured lead are installed internally at any seam areas of the compartment, and reach rods are provided from remote operation of valves.

The sample is directed into the shielded portable cask so that a nominal 62-mi sample can be safely collected and transported to the analysis apparatus. Streaming is minimized by offsetting the openings in the cask from the direct line of sample and by use of a shielded cap during transport. Dose is minimized procedurally by use of a special hand tool to disconnect the cask from the system.

After the sample is collected in the cask, it is transported and positioned under the analysis equipment behind a shielded door. The cask is connected to the analysis system via flexible tubing of minimum volume, and the sample is pumped into a modified closed loop gas expansion rig. The cask with connections, in fact, forms part of the closed loop for gas expansion. The gas expansion rig has been modified to fit into a shielded box 50cm x 35cm x 35cm ID. The box is steel-encased lead, with a minimum thickness of 3 in., with a lead glass viewing window. The apparatus uses all solennoid-operated valves to preclude high personnel exposure. The box is ventilated into the Primary Auxiliary Building's ventilation system via an in-line vaneaxial fan.

The sample is recirculated through the gas expansion rig, which contains 30 mls of demineralized water for analysis of dissolved total gas and hydrogen. A gas sample is withdrawn by syringe through a septum for hydrogen analysis by gas chromatography. A second gas sample is withdrawn and injected into a glass bulb for isotopic analysis on the gamma spectroscopy system. On completion of the gas analyses, the diluted (1:1.5) degassed liquid is pumped to a beaker within the shielded box. Samples of the diluted liquid in the breaker are withdrawn for liquid radioisotopic, boron and chloride analyses. An undiluted sample can be obtained within 24 hours and analyzed for chlorides within 30 days. The radioisotopic and boron samples may be diluted again as necessary to limit personnel dose. On completion of all analyses, the sample left in the beaker is pumped to a waste cask for disposal. The system can then be flushed with water. The capabilities are listed in Table 9.4-4.

The primary method of pH measurement would be the use of an inline sensor mounted in a shielded cask. The sample is obtained in the same manner as the initial pressurized reactor coolant sample. A pH measurement can be obtained by taking a second sample and performing a pH analysis with the analysis system prepared without dilution water. The undiluted, second sample would be pumped into a beaker within the shielded box containing a pH electrode. On completion of the analysis, the sample is pumped to a waste cask for disposal. The system can then be flushed with water.

9.4.2.2 Containment Atmosphere and Plant Vent Post – Accident Sampling System

The post-accident containment atmosphere and plant vent sampling system is designed to obtain representative samples of the containment air and stack for isotopic analysis. The system is designed to be utilized when the normal sampling system is inaccessible during periods of abnormally high release rates.

The containment atmosphere and plant vent sampling lines are electrically heat traced in order to prevent moisture condensation. The heat trace cable provided is of the self-regulating type selected to maintain the containment atmosphere sample line at 250°F and the plant vent stack sample line at 120°F. Two thermostats are provided per heat tracing zone: one for temperature regulation, the other for a low temperature alarm.

The basic system consists of containment air and stack sample shielded compartments located on the 41 foot elevation of the Primary Auxiliary Building. The containment air sample system consists of three lead shielded compartments with hinged doors. One compartment contains the sample collection cartridges for iodine and particulate. Another compartment houses the gas sampling cylinder and minimal tubing. The third compartment houses most associated tubing and a sample pump. The plant vent system is similar but all tubing, sample media and a pump are housed in one shielded compartment.

A noble gas sample is withdrawn by syringe through a port in the shielded compartment and analyzed for hydrogen by gas chromatography and for activity by gamma spectroscopy. A silver zeolite cartridge and a millifilter are used to collect iodine and particulate samples. Bottled gas is purged through the system after collecting the sample to reduce personal exposure during removal and transport of the sample media. The samples are then analyzed for iodine and particulate on the gamma spectrometer.

If this sampling capability (sample line common with retired monitor R-13) is determined to be inoperable when the reactor is in MODES 1, 2, 3 or 4, then it shall be restored to operable status within 72 hours or perform the following:

a) initiate a pre-planned alternate sampling / monitoring capability as soon as practical but no later than 72 hours after identification of failures. If the capability is not restored to operable status within 7 days, then,

b) submit a special report to the USNRC within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system.

In addition, the sample line common with retired monitor R-13 shall be tested every 18 months to ensure a representative sample can be drawn. This test is governed by Plant Procedures.

9.4.2.3 Main Steam Post-Accident Sampling System

The post-accident main steam sampling system is designed to collect a condensed liquid sample of the main steam during accident conditions to verify steam generator integrity.

Upstream isolation valves are opened on the steam generator side of the main steam isolation valves. The normal operation sample root valve is shut and the accident cross-connect valve is opened. A sample is obtained from the main steam sample sink isolation valve in the secondary laboratory on the 15 foot elevation of the turbine building.

An isotopic analysis of the main steam sample is performed using the gamma spectroscopy system.

Design Codes and Criteria

The general controlling standard for the design, fabrication, installation and testing of the Fuel Storage Building fuel cask crane is the Electric Overhead Crane Institute (EOCI), Inc., Specification No. 61.

The crane, bridge, trolley and hoist are controlled by EOCI No. 61. Specifications for the following organizations are referenced therein:

1) American Gear Manufacturers Association
2) American Institute of Steel Construction
3) Association of Iron and Steel Engineers
4) American National Standards Institute
5) American Society for Testing and Material
6) American Welding Society
7) National Electrical Code, National Fire Protection Association
8) National Electric Manufacturers Association

The crane rail and structural steel supporting structures were controlled by the American Institute of Steel Construction, "Manual of Steel Construction" 1964. All structural steel is ASTM A-36. The crane rail is in accordance with Manufacturers Standards and ASTM A-1. Design of the cables was controlled by the EOCI No. 61 code. In addition, the following are applicable to cables:

1) RRW-40-C, which is a Federal Standard for wire rope representing the industry standard

2) ANSI B30.2.0 for Overhead and Gantry Cranes.

The lifting hooks were purchased, fabricated and load tested to manufacturers standards. The hoods were ultrasonically tested to detect any flaws.

The hooks are inspected, tested, and maintained in accordance with ANSI B30.2-1976. Overhead and Gantry Cranes. When the crane hooks are inactive for a period of time longer than a specified inspection, test, or maintenance frequency, the inspection, test, or maintenance activity should be performed prior to their use. (NCR letter dated February 13, 1985, Control of Heavy Loads (Phase 1)

The manipulator crane structure was designed in accordance with Electric Overhead Crane Specification No. 61. The design load was specified as 5626 lbs (the weight of three fuel assemblies at 1575 lbs each plus the weight of the gripper tube at 900 lbs). All loading supporting members were designed with a 5 to 1 safety factor on this design load. The crane was pre-operational load tested with 6000 lbs. Normal operating load for the crane is 2475 lbs although emergency procedures for removing stuck fuel assemblies may require occasional loading up to 6000 lbs. Seismic loads used for design were based on 0.5 g horizontal and 1.0 g vertical accelerations which are greater than the accelerations at the installed location. To resist design basis earthquake forces, the equipment was designed to limit the stress in the load bearing parts to 0.9 times the ultimate stress for a combination of normal working load plus design basis earthquake forces.

The structural material for the spent fuel pit bridge and hoist was designed to ASTM-A373. The design load was specified as 2000 lbs on the hoist and 250 lbs per square foot on the bridge. All load supporting members were designed with a 5 to 1 safety factor at these design loads. The hoist was pre-operational load tested with 2500 lbs while the normal operating load will be 1750 lbs (fuel assembly plus fuel handling tool). Seismic loads used for design were based on 0.5 g horizontal and 1.0 g vertical accelerations, which are greater than the accelerations at the installed location. To resist design basis earthquake forces, the equipment was designed to limit the stress in the load bearing parts to 0.9 times the ultimate stress for a combination of normal working load plus design basis earthquake forces.

For seismic considerations of the Fuel Storage Building spent fuel cask crane, the fuel crane bridge was evaluated to determine the potential for the trolley to lift off the crane bridge rails or for the crane bridge to lift off its track support in the event of a seismic disturbance. The crane bridge and trolley were analyzed for the design basis earthquake both loaded and unloaded for various positions of the trolley using response spectra modal analysis with 2% damping. It was determined that the downward force due to gravity exceeds the maximum upward seismic wheel load due to combined vertical and horizontal accelerations by a factor of 1.2.

As this is the only potential for bridge or trolley lift-off or overturning, no potential hazard exists to any seismic Class 1 function, and vertical restraints were not required.

The wheels of the bridge and trolley are shaped such that sliding perpendicular to the rail would not be possible. The lateral load from an earthquake on the trolley crane rail is about 50% greater than the lateral loads from impact that the AISC Code specifies for design within working stress limits. The stresses on the crane rail are low due to the earthquake load. For this reason, no failure of the crane rail is anticipated. The design load rating of this crane is anticipated. The design load rating of this crane is 40 tons with a 5 ton auxiliary hook.

Other pre-operational test loads on components of the FHS were:

a) 125% of design rating on spent fuel cask crane

b) Functional check-out for operability using a dummy fuel assembly that approximates 100% of the operational load to be handled.

Test loads used throughout the plant life shall be equal to or greater than the maximum load to be assumed by the hoist crane during refueling operations.

9.5.2 System Design and Operation

The reactor is refueled with equipment designed to handle the spent fuel underwater from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Boric acid is added to the water to ensure subcritical conditions during refueling.

The Fuel Handling System may be divided into two areas: the reactor cavity, which is flooded only during plant shutdown for refueling, and the spent fuel pit, which is kept full of water and is always accessible to operating personnel. These two areas are connected by the Fuel Transfer System consisting of an underwater conveyor that carries the fuel through an opening in the plant containment. (See Figure 9.5-1)

The reactor cavity is flooded with borated water from the Refueling Water Storage Tank. In the reactor cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the fuel transfer cart by a manipulator crane. In the spent fuel pit, the fuel is removed from the transfer system and placed in storage racks with a long manual tool suspended from an overhead hoist. The fuel can be removed from storage and loaded into a shipping cask for shipment from the site.

New fuel assemblies are received and stored in racks in the new fuel storage area. The new fuel storage area is sized for storage of the fuel assemblies and control rods normally associated with the replacement of 72 fuel assemblies.

Major Structures, Systems and Components Required for Fuel Handling

Reactor Cavity

The reactor cavity is a reinforced concrete structure that forms a pool above the reactor when it is filled with borated water for refueling. The cavity is filled to a depth that limits the radiation at the surface of the water to 2.0 milliroentgens per hour during fuel assembly transfer.

The cavity is large enough to provide storage space for the reactor upper and lower internals, the control cluster drive shafts, and miscellaneous refueling tools. The floor and sides of the reactor cavity are lined with stainless steel.

The reactor vessel flange is sealed to the cylindrical side walls of the reactor refueling cavity by a Presray seal. This seal design utilizes gas pressurization to inflate and uniformly compress an oval cross-section reinforced synthetic rubber (EPDW) envelope structure to effect a seal. The required inflation pressure is specified at 31 psig (equivalent head of water to be sealed plus 20 psig). This seal is installed and inflated after reactor cooldown but prior to flooding the cavity for refueling operations.

The wedge shape at the top of the device is designed to effect a pressure tight seal by virtue of the hydrostatic head of water even if pneumatic inflation pressure were to be lost.

The Presray seal design includes two independent gas inflation connections that are in simultaneous use during service. Each gas connection point at the seal is equipped with a fixed orifice device that limits seal deflation to a minimum of 10 psig should either one of the inflation sources malfunction. Both inflation sources are monitored by on-line "air supply to seal" pressure gauges. The difference in indicated delivery pressure will indicate a gas source malfunction or the failure of a gas connection. Any inflation gas leak into the refueling pool volume will be revealed by gas bubbles appearing at the surface of the pool.

Reactor Cavity Filtration System

The Reactor Cavity Filtration System provides the capability of filtering the water in the reactor cavity whenever the cavity is filled. This filtration system maintains water clarity and removes suspended radioactive particles.

The system consists of a skid carrying four stainless steel filter units, associated piping, and valving mounted the 95' floor elevation against the northwest face of the shielding around Steam Generator No. 33. It is enclosed on its three exposed sides by shielding. A separate skid supports a motor driven centrifugal pump. Flexible hoses and hard piping connects both suction and discharge of the filter package with the refueling canal. All surfaces in contact with refueling water are either stainless steel or synthetic hose and filter medium. At present, the Reactor Cavity Filtration system is partially disassembled and is not used. All disassembled equipment can be reinstalled at a future date if it is desired to use the system. Filtration of the reactor cavity water can be performed using the Spent Fuel Pool Cooling System or a temporary augmented system, as described below.

For a discussion of spent fuel pit dewatering as a result of a tornado, refer to Section 16.4.

The ventilation system in the Fuel Storage Building enclosing the spent fuel pit was designed so that there is a slight negative pressure inside the building during normal refueling operations. Whenever the ventilation system is required to be in operation, the bypass dampers around the charcoal filter must be manually closed and leak tested to assure that it is properly sealed. On a high radiation alarm, the following actions automatically take place:

1) Building ventilation supply fans are secured,

2) Dampers at ventilation supply fans close,

3) If open, rolling door closes,

4) inflatable seals on main doors and truck doors are actuated (R-5 operability does not require this function, however), and

5) Exhaust fan continues to run.

Under these conditions, the maximum calculated in-leakage to the building (caused by the operation of the exhaust fan) would be 20,000 cfm with a one-half inch of water negative pressure inside the building. Thus, following the release of radioactivity in the Fuel Storage Building, there will be zero air leakage from the building proper, and the entire exhaust from the building will pass through roughing, HEPA, and charcoal filters before going up the plant vent.

A pushbutton switch is provided adjacent to the 95' elevation door leading to the Fan House. This switch allows the Fuel Storage Building Exhaust Fan to be momentarily shut down and air removed from the door seal thereby allowing the door to be opened. The fan will automatically restart and the door is resealed after a preset time has elapsed (approximately 30 seconds).

The handling of irradiated fuel in the Fuel Storage Building or movement of the spent fuel cask or cask crane over the Spent fuel Pit are prohibited when the fuel storage building emergency ventilation system is inoperable.

Since the spent fuel cask must be loaded in the spent fuel pit, the crane must carry a heavy load, namely the cask, over the pit. The bases for the acceptability of this design are:

1) During normal fuel handling operations, heavy loads (above 2000 lbs) cannot be carried over spent fuel, and

2) Even in the event that the spent fuel cask is dropped over the pit, the loss of water from the resulting failed liner plate and cracked concrete base is inconsequential.

Loss of water in the spent fuel pit and the resultant uncovering of the spent fuel by way of drains and permanently connected system cannot take place for the following reasons:

1) The suction of the spent fuel pit pump is taken from a point approximately six (6) feet below the surface of the pool; therefore this pump cannot be used to uncover the fuel, even accidentally.

2) The spent fuel pit pump discharges into the pool approximately seven (7) feet above the top of the spent fuel assemblies; therefore this pipe could not accidentally become a siphon to uncover the fuel.

3) The skimmer pump takes suction from, and discharges to the surface of the pool; therefore it could not accidentally or otherwise uncover the spent fuel.

4) There are no drains on the bottom or side walls of the spent fuel pit. Draining would have to be done deliberately by a temporary pump.

5) The spent fuel pit cooling loop was designed to seismic Class II and the cleanup equipment loop was designed to seismic Class III criteria; however, their failure could not result in the uncovering of the spent fuel, as explained above.

A radiation monitor (R-5) is located in the Fuel Storage Building. The monitor provides continuous indication of the radiation level with high radiation alarm given both locally and in the Control Room. The air filtration system for the Fuel Storage Building is automatically actuated on high radiation signal. Radiation levels in the spent fuel storage area shall be monitored continuously whenever there is irradiated fuel stored therein. If the monitor is inoperable, a portable monitor may be used. Equipment is also provided to monitor spent fuel pit water level with low level annunciated in the Control Room. (See Section 11.2)

The primary source of makeup water to the spent fuel pit is the Primary Makeup Water Storage Tank, which is a seismic Class I component. The pumps and most of the piping associated with this tank are also seismic Class I. The makeup water loop to the spent fuel pit is seismic Class II, as is the spent fuel pit cooling loop. The cleanup equipment and skimmer loops are seismic Class III. Additional backup can be provided through a temporary connection from the plant demineralizers or from the Fire Water Tank.

In addition, there is a second spent fuel pool cooling system pump to provide standby capacity. There is also a provision for adding a portable cooling pump.

Storage racks provided to hold spent fuel assemblies were erected on the pit floor. The racks were designed so that it is impossible to insert fuel assemblies in other than the prescribed locations (there is insufficient space between the rack assembly and the SFP wall), thereby ensuring the necessary spacing between assemblies. Control rod clusters or other inserts are stored inside the spent fuel assemblies.

Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing and the manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper in the event of a maximum potential earthquake.

Only core components or tools required for the placement or removal of core components are handled over an open reactor vessel.

Any time the reactor vessel is open, the following precautions are taken to assure that foreign materials do not inadvertently get into the reactor vessel:

1) All personnel tape the cuffs, pockets, buttons, etc. of their "anti-contamination clothing"
2) A barrier is established surrounding the reactor cavity to prevent unnecessary movement in the area
3) Only the minimum number of people required to safely perform the job are allowed in the area
4) All facets of the Quality Assurance Plan are in effect and enforced
5) The cranes are visually inspected before the reactor vessel is opened
6) Prior to lifting of the head, an NDT of the hook is performed.

All fuel handling operations, including core alterations, are performed under the supervision of an individual holding either a Senior Reactor Operator license or a Senior Reactor Operator license limited to fuel handling, as established in 10 CFR 50.54 (m) (2).

Discussions of the effects of the seismic Class III Fuel Storage Building crane on seismic Class I functions are found in Section 16.4

The manipulator crane bridge and trolley are restrained on the rails by the following means: (See Figure 9.5-3)

1) Horizontally – by guide rollers (cam follower) at each wheel on one truck only. The rollers are attached to the bridge truck at the wheels and contact the vertical faces of the rail to prevent horizontal movement.
2) Vertically – by anti-rotation bars, in the vicinity of each wheel at all 4 wheel locations. The anti-rotation bars are carbon steel bars bolted to the truck and extending under the rail flange, to prevent lifting of any wheel from the rail.

Polar Crane

The containment building polar crane is utilized to remove and replace heavy loads during refueling operations. These include:

1) Control rod drive missile shield

2) Reactor vessel head

3) Reactor internals

All standard modes of failure were considered in the design of the polar crane. These modes of failure were provided for by utilization of a minimum safety factor of 5 based on the ultimate strength of the material used in the design of cables, shafts and keys, gear teeth and brakes. All crane equipment was sized to handle the single heaviest load realized during plant operation. All lifts are made by qualified personnel. The equipment is properly maintained and periodically inspected by qualified personnel. An analysis of impact loading on the reactor vessel due to dropping the reactor vessel head is provided in Section 9.5.3.

Fuel Storage Building Crane

The fuel storage building crane is utilized to move loads not exceeding 2000 lbs during normal refueling operations. These loads include:

1) Irradiated specimens

2) Neutron source

3) Crane load block

4) Burnable poison rod and handling tool.

Mechanical stops incorporated on the bridge rails of the fuel storage building crane make it impossible for the bridge of the crane to travel further north than a point directly over the spot in the spent fuel pit that is reserved for the spent fuel cask. Therefore, it will be impossible to carry any object over the spent fuel storage areas north of the spot in the pit that is reserved for the cask with either the 40 or 5-ton hook of the fuel storage building crane. It is possible to use the fuel storage building crane to carry objects over the spent fuel storage areas that are directly east of the spot in the pit that is reserved for the cask. However, the technical specifications and plant procedures prevent any object weighing more than 2,000 pounds from being moved over any region of the spent fuel pit when the pit contains spent fuel. Therefore, the storage areas directly east of the spot in the pit that is reserved for the cask are protected from heavy load handling by administrative controls.

All standard modes of failure have been considered in the design of the fuel storage building crane. These modes of failure were provided for by utilization of a minimum safety factor of 5 based on the ultimate strength of the material used in the design of cables, shafts and keys, gear teeth and brakes.

All crane equipment was sized to handle the single heaviest load realized during plant operation. All lifts are made by qualified personnel. The equipment is properly maintained and periodically inspected by qualified personnel. An analysis of impact loading of the spent fuel cask into the spent fuel storage pool is provided in Section 9.5.3

Spent Fuel Pit Bridge

The spent fuel pit bridge is a wheel-mounted walkway spanning the spent fuel pit that carries an electric monorail hoist on an overhead structure. Fuel assemblies or inserts are moved within the spent fuel pit by means of a longhandled tool suspended from the hoist. The hoist travel and tool length were designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

Fuel Transfer System

The fuel transfer system, shown in Figure 9.5-1, is a motor winch driven conveyor car that runs on tracks extending from the refueling cavity through the transfer tube and into the spent fuel pit. The conveyor car received a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is lowered to a horizontal position for passage through the tube, and then is raised to a vertical position in the spent fuel pit.

During plant operation, the conveyor car is stored in the refueling canal. A blind flange is bolted on the transfer tube to seal the Reactor Containment.

Rod Cluster Control Changing Fixture

A fixture is mounted on the reactor cavity wall for removing rod cluster control (RCC) elements from spent fuel assemblies and inserting them into new fuel assemblies. The fixture consists of two main components: a guide tube mounted to the wall for containing and guiding the RCC element, and a wheel-mounted carriage for holding the fuel assemblies and positioning fuel assemblies under the guide tube. The guide tube contains a pneumatic gripper on a winch that grips the RCC element and lifts it out of the fuel assembly. By repositioning the carriage, a new fuel assembly is brought under the guide tube and the gripper lowers the RCC element and releases it. The manipulator crane loads and removes the fuel assemblies into and out of the carriage.

Refueling Procedure

Refueling requirements and procedures are contained in the Technical Specifications and in this FSAR section.

Preparation

a) The reactor is shut down, cooled to T_{avg}, $\leq$ 140 ^{o}F and boron concentration is checked

b) A radiation survey is made and the containment vessel is entered

c) The control rod drive mechanism (CRDM) missile shield is removed to storage

If sampling shows the presence of fission products indicative of a cladding failure, the sampling lines are closed off by valves on the can and the fuel assembly is removed to the spent fuel storage racks to await shipment. Design of the failed fuel test cans complies with 10 CFR 72.

Failed fuel can also be detected through the use of the in-mast sipping system (which is essentially a version of the sipping can that is permanently attached to the fuel handling equipment) and the poolside ultrasonic failed fuel detector, which uses a probe to examine each fuel rod for entrained water.

Drop of Spent Fuel Element Cask Into Spent Fuel Storage Pool

As indicated in Section 9.5-2, the cask cannot be transported above fuel assemblies, hence, under no circumstances can the fuel assemblies be in jeopardy from the cask. However, the event that the cask would drop into the pool has been analyzed; the basic assumptions for analysis were as follows:

a) The drop would be from the cask's highest position which is 5 feet above the water surface and 43 feet above the bottom of the pool

b) The cask is fully loaded and weight 40 tons.

The results of the analysis indicate that the cask would hit the bottom of the pit with a velocity of approximately 40 ft/sec, assuming a conservative drag coefficient of 0.5. In comparison, the cask would have reached a velocity of 52 ft/sec if dropped through 43 feet in the air.

Using the Ballistic Research Laboratories formula for the penetration of missiles in steel, the depth of penetration of the cask into the 1-inch wear plate covering the ½-inch pit liner plate would be 0.32-inch, assuming the cask struck the wear plate while in a perfectly vertical position. In the event that the cask falls through the water at an angle, its terminal velocity would be somewhat less because of the increased drag. However, the cask would strike the wear plate with an initial line contact and would penetrate the wear plate and the pit liner plate, causing some cracking of the concrete below. This reinforced concrete is a minimum of 3'-7" thick and rests on solid rock.

Water would initially flow through the punctured liner plate and fill the cracks in the concrete. As the pit is founded on solid rock and much of the bottom of the pit is below the surrounding grade, very little water can be lost from the pit. The capacity of the makeup demineralized water supply to the pit is 150 gpm. In addition, the spent fuel pit cooling system piping includes a 4" blind flange connection for temporary cooling.

Since the bottom of the spent fuel pit is an average 24 feet below grade and since no equipment areas are in the vicinity, there can be no flooding of other areas outside the Fuel Storage Building and subsequent damage to equipment.

Siding Panel as a Missile

Analysis has been made for the drop of a 32-1/2 ft long by 19 ft wide by 2 in thick insulated siding panel missile weighing 1860 lbs through 50 ft of free fall onto the water surface of the spent fuel pit. Although such a missile would logically be expected to plane in the water and impact the side walls of the storage pool, the analysis shows that even with the highly conservative assumption that it penetrates the water in a guillotine fashion, such that drag is

based on the 19 ft x 2 in minimum cross sectional area, the drag and buoyancy forces prevent fuel damage.

The missile kinetic energy required to damage the fuel assembly cladding if 6900 ft-lbs. The missile kinetic energy variation with water depth is computed from:

$$D(KE)/dy = W - (\zeta/2g)\,(2KE)\,(W/g)^{-1} C_D A - \zeta A_y$$

where KE = missile kinetic energy
W = missile weight
ζ = missile cross sectional area = 3.2 ft^2
z = water density = 62.4 lb/ft^3
C_D = drag coefficient = 1.0
g = gravity constant – 32.2 ft/sec^2
y = depth of water penetrated

Since the fuel storage pool is 40 ft deep with an excess of 23 ft of water over the top of the fuel assemblies, the postulated missile will be buoyed up before it strikes the fuel storage racks and fuel. Should it be postulated that tornado winds reduced the water level by 6 ft, the missile would impact the storage cell, with a striking impact energy of 2875 ft-lb per cell. However, there would be substantial margin to fuel clad failure.

The equivalent analysis was made for a 12-ft x 12-in x 4-in wooden plant striking the water vertically with a velocity of 90-mph. After 23 ft of water penetration, the plank kinetic energy is 4784 ft-lbs under the minimum drag area assumption. This would be insufficient to cause fuel failure even if the plant were to miss the storage rack and impact the top end of a stored fuel assembly. In order to illustrate the effect of planing, which would result from unsymmetrical impact of this missile on the water surface, a three dimensional model was analyzed.

Figure 9.5-4 shows calculated motion when the missile axis and missile velocity are in alignment with each other but impact is 5 degrees of vertical. Figure 9.5-5 shows the motion when the missile axis and initial velocity are misaligned by 5 degrees. It is seen that quite small deviations from a perfectly symmetrical water impact results in rotation of the missile, significantly reducing its penetration depth.

The effect of an automobile weighting 4000 lbs, entering the pool at 17 mph with 25 sq. ft contact area, was also analyzed. Due to the slower speed and the much larger drag area, the automobile will have an impact energy of 3133 ft-lb per cell, far less than the energy required for clad damage (see NSE 00-3-039 SFPC for details on missile analysis).

It is concluded that the storage pool water and storage cells provide effective protection against tornado missiles and that the chance of fuel damage by such missiles is low. Considering this, the low probability of a strong tornado striking the site, the fact that radioactive iodine in a stored fuel assembly is less than 10% of the shutdown value except of 6% of the year (first 2.3 half-lives), and the unstable and dispersive meteorological conditions accompanying a tornado, further protection is not needed.

Uplift for CRS

The individual spent fuel racks were not designed to withstand uplift forces, as a force applied to one cell will be distributed to the entire array of fuel cells (racks) in the pit. This occurs because the racks are not attached to the bottom liner of the pit but rather are interconnected.

The dead weight of adjacent fuel cells (racks) and fuel elements relieves any uplift force on the rack support members.

No force of significant magnitude can be applied to the fuel racks when removing a fuel assembly as the inside face of each rack opening was fabricated free of all burrs and rough edges and is smooth and clean.

Reactor Impact Loads

The worst case of impact loading, that of dropping the reactor vessel head onto the reactor vessel was evaluated. A weight of 300,000 lbs, was assumed to drop from the maximum possible height of 54 feet above the reactor vessel flange. The total impact load was calculated to be 74.5×10^6 lbs or 18.6×10^6 lbs acting on each vessel support.

Using this load, the maximum stress at the nozzle-to-shell juncture occurs on the outlet nozzle and is 62,600 psi. This compares favorable with the allowable stress of 84,000 psi (faulted condition). The maximum calculated direct shear stress on the nozzles of 14,300 psi, which is less than the allowable of 33, 600 psi (faulted condition). The shear stress calculated by driving the nozzle support pad through the nozzle is 28,000 psi, which is less than the yield stress (44,500 psi).

All stresses are within allowable limits, and therefore, the structural integrity of the nozzle and nozzle supports is maintained.

No reduction in dynamic load was taken due to the dampening effect of the head falling through 23 feet of water.

Therefore, it is considered very unlikely that the dropping of the reactor vessel head (or core barrel assembly or missile shield) will disrupt the flow of coolant to and from the reactor vessel and refueling canal. Some local yielding of the nozzles (supports) may occur that could cause relative displacement between the vessel seal ledge and concrete seal support ring that could cause water seal failure. However, loss of all refueling water down through the annulus between vessel and concrete is not a safety-related item. Any water lost in this manner would merely drain to the containment sump.

9.5.4 Minimum Operating Conditions

Minimum operating conditions are specified in the Technical Specifications, FSAR Sections 1.3.6, 9.5.2 and 9.5.5, and plant procedures.

9.5.5 Tests and Inspections

Upon completion of core loading and installation of the reactor vessel head, certain mechanical and electrical tests were performed prior to initial criticality. The electrical wiring for the rod drive circuits, the rod position indicators, the reactor trip circuits, and the incore thermocouples, were tested at the time of installation. The tests were repeated on these electrical items before initial plant operation.

The following tests shall be performed on cranes or hoists utilized in irradiated fuel movement prior to fuel movement:

- Dead-load test (weight must be equal to or greater than the maximum load assumed by crane or hoist).
- Thorough visual inspection after dead-load test.
- Test of interlocks and overload cutoff devices, prior to movement of fore components.

9.5.6 Crane Operator Qualifications

A qualification program for crane operators was established requiring the following:

a) Certification by a company physician that the crane operator met the physical standards set forth in USAS B30.2.0-1967, Article 2.3.1.2

b) Successful completion of an oral and practical examination given by a designated experienced Crane Instructor

c) Certification by the crane operator that he has read, understands and will comply with the operational and safety requirements set forth by OSHA and USAS B30.2.0-1967.

This qualification program meets the requirements of Chapter 2-3 of ANSI B 30.2-1967, "Operation – Overhead and Gantry Cranes", as developed by the American National Safety Code for Cranes, Derricks, Hoists, Jacks and Slings.

The crane operator qualification program is one component of a defense-in-depth approach used to properly manage the lifting of heavy loads near spent fuel and safe shutdown systems in accordance with NUREG-061; "Control of Heavy Loads at Nuclear Power Plants," dated July 1980. Other components include use of safe load paths and load handling procedures. In addition, affected cranes and lifting devices are subject to industry standards for design, testing, inspection and maintenance. Specific requirements stated in the Authority's responses to NUREG-0612 are summarized in the NRC's Safety Evaluation Report dated February 13, 1985.

Ch.

10

10.2 SYSTEM DESIGN AND OPERATION

The Steam and Power Conversion System Process Flow Diagram is shown on Figure 10.2-1. Process and Instrumentation Diagrams for the following systems: Main Steam; Condensate and Boiler Feed Pump Suction; Condensate Polishing; Boiler Feedwater; Extraction Steam; Heater Drains and Vents; Moisture Separator and Reheater Drains and Vents; Condenser Air Removal and Water Box Priming; Circulating Water; Chemical Feed,; Auxiliary Steam and Condensate for Nuclear Equipment; Auxiliary Steam Supply and Condensate Return; and Steam Generator Blowdown and Blowdown Recovery Systems are shown on Figures 10.2-2 through 10.2-13, and 10.2-47 and 48.

Heat Balance diagrams at loads of 1,068,701 kW(e); 1,022,793 kW(e); 1,000,639 kW(e); 766,350 kW(e); 510,897 kW(e); and 255,448 kW(e) are shown on Figures 10.2-14 through 10.2-19.*

10.2.1 Main Steam System

The Main Steam System conducts steam in a 28 inch pipe from each of the four steam generators within the reactor containment through a swing disc type isolation valve and a swing disc type non-return valve to the turbine stop and control valves. The isolation and non-return valves are located outside of the Containment. The four lines are interconnected near the turbine. The design pressure of this system is 1085 psig at 600 F. A steam flow-meter (flow venturi) is provided upstream of the isolation and non-return valves in the line from each steam generator to measure steam flow. Steam flow signals are used by the automatic feed-water flow control system (see Chapter 7). The flow venturi also limits the steam flow rate in the event of a steam line break downstream of the venturi. Steam pressure is measured upstream of the isolation and non-return valves.

The isolation valves contain free swinging discs that are normally held out of the main steam flow path by an air piston. The isolation valves are designed to close in less than five seconds. Air receiver tanks are provided in the instrument air piping to the Main Steam Isolation Valves (MSIV) operators to compensate for pressure transients in the instrument air system in order to prevent spurious MSIV closure. The isolation valves are automatically closed (closure of isolation valves initiates unit trip) on receipt of the following signals from the steam line break protection system:

1) High steam flow in any two out of the four steam lines, coincident with low steam line pressure or low T_{avg}; or

2) Two sets of the two-of-three high-high containment pressure signals; or

3) Manual actuation (one at a time).

* These figures are based on original plant equipment and are provided for historical purposes only.

Tests on the main steam isolation valves and the non-return valves performed by the manufacturer were:

1) Tight seating at 1200 psi maximum and 50 psi minimum differential pressures

2) Stem gland leakage not to exceed 1 cc of water per hour per inch of stem diameter when subjected to a hydrostatic leak test pressure of 1100 psig, or not to exceed 0.03 SCFH of air per inch of stem diameter with a differential pressure of 80 psi

3) Non-destructive testing.

The basis for the steam generator stop (stop-check) valves and the steam generator check valves design leakage rates was the Manufacturer's Standardization Society Specification MSS-SP-61 and standard industry criteria for steam leakage rates. The acceptance criteria for shop tests and stem leakage rates were that steam valve glands be designed and packed so that leakage along the stem does not exceed 1 cc of water per hour per inch of stem diameter when subjected to a hydrostatic leak test pressure of 1100 psig, or 0.03 SCFH of air per inch of stem diameter with a differential pressure of 80 psig. The basis for this criterion was sound, up-to-date engineering practice, and was acceptable to valve manufacturers and used by them in the design and fabrication of their products.

The air operated main steam line stop (stop-check) valves are tested at refueling intervals to verify their ability to close within the specified time upon receipt of a closure signal. See Section 3.7.2 of the Technical Specifications.

Testing of main steam isolation valves under steam flow conditions is not justified. This testing would cause a severe transient and the collapse of steam bubbles in the steam generator shell side resulting in a low-low water level plant trip. Also, because of the valve design, the valves cannot be reopened against any differential pressure across the valve. The valve operator was designed only to close the valve during steam flow conditions. Thus a valve test at steam flow conditions would necessitate bringing the plant to shutdown conditions after each valve test.

The main steam non-return valve, as any swing type check valve, closes upon reverse flow of steam in case of accidental pressure reduction in any steam generator or its piping.

Each steam line is provided with a venturi-type restrictor. The flow restrictors are designed to increase the margin to Departure from Nucleate Boiling (DNB), and thereby reduce fuel clad damage by limiting steam flow rate consequent to a steam line rupture and thereby reducing the cooldown of the primary system.

1) Failure in the single supply line to the auxiliary feedwater pump turbine anywhere downstream of the stop check valves

2) Failure of the auxiliary feedwater pump pressure relief valve at low pressure in the steam generator

3) Closure of the two automatic isolation valves from a high temperature signal in the building

The system design ensures a steam supply to the auxiliary feedwater pump turbine in the event of:

1) Failure of one main steam line upstream of the stop check valve. The stop check valves prevent flow to the broken main.

2) Failure of the pressure reducing valve. This valve fails open on loss of control (either through the loss of electrical power or air supply); the turbine pressure relief valve ensures that sufficient steam flows to the turbine while maintaining a safe pressure at the turbine. This applies at safety valve set pressure plus accumulation in the steam generator.

Protection of the Main Steam System has been provided in the event of a failure downstream of the stop check valves by limiting the take off points at the steam main to 3-inch nominal pipe size. This restricts the consequences of the rupture of this pipe to a release of steam less severe than that resulting from the sticking open of a safety valve.

Protection of the AFS from lack of water to pump suctions is provided by operator action. If it is discovered that one or both of the valves in the single auxiliary feedwater supply from the CST are closed in MODES 1, 2, or 3, then the AFS is immediately places in manual mode. The AFS is returned to automatic mode once a water supply has been restored.

Feedwater is supplied by the steam turbine driven pump to all four steam generators through individual feedwater regulating valves that are controlled either from the main control board or locally at the valves. The drive unit is a single stage turbine, capable of quick starts from cold standby, and is connected directly to the pump. This turbine is started by opening the pressure reducing valve between the turbine supply steam header and the main steam lines. The turbine sleeve journal bearings are ring oil lubricated water cooled. The pump uses oil slinger lubricated ball bearings.

The motor driven pump loop utilizes two pumps with ring lubricated ball bearings. Each pump has a design capacity of 400 gpm, but conservatively assumed to be 340 gpm (343 gpm for Loss of Normal Feed/Loss of AC Power events), and their discharge piping is arranged so that each pump supplies two steam generators. The design performance characteristics for these pumps is given in Figure 10.2-36.

One motor driven pumps has sufficient capacity to maintain a sufficient water inventory in the steam generators to which it is connected, preventing relief through the primary coolant system pressurizer relief valve following a reactor/turbine trip. Thus, in the remote chance of loss of the steam driven pump, a single motor driven pump is adequate to ensure safety of the public.

The motors are of open drip-proof design. In the event of complete loss of power, electrical power is restored automatically from the diesel generators.

The three auxiliary feedwater pumps are located in an enclosed room in the Auxiliary Feedwater Building that houses the area of the main steam and feedwater penetration, immediately outside of the Reactor containment (see Figures 10.2-38 and 10-2-30). The distribution piping is seismic Class I throughout. The piping was designed to ensure that a single failure will not compromise the system function.

All components within the AFS boundary were designed to seismic Class I criteria, as noted in Section 16.1. This includes designing for the Design Basis Earthquake and pipe breaks (pipe whip). The AFS has tornado protected pumps and redundant water supplies, as discussed in Section 16.2.

The possibility of internally generated missiles from the auxiliary feedwater steam driven pump turbine has been evaluated. The evaluation findings note that missiles generated at destructive overspeed could penetrate the turbine casing and that there are possible targets that require protection from such a missile. In view of this, a shield around this turbine was designed. The system is otherwise protected from a main turbine missile by incorporation of redundant water supplies, missile protected pumps and redundant, separate pipes feeding the four steam generators.

Protection of the AFS from excessive vibration and overheating is provided by means of a 2-in recirculation line without flow restricting fixed orifices. Pressure reducing control valves, shut-off isolation valves and check valves were sized to minimize vibration and pipes were routed to minimize bends. The control valves were designed to fail open.

Single Failure Criteria

Redundant auxiliary feedwater supply is provided by using two pumping systems with independent motive power sources. In the event of a complete loss of offsite power, electrical power to the motor driven pumps is supplied by the diesel generators. The turbine driven pump is completely independent of the motor driven pumps and there are redundant power supplies to the motor driven pumps.

The three auxiliary feedwater pumps can be started remotely-manually from the Control Room or locally at the pump room. Thus, provision exists for manual initiation on the component level, but no such provisions exist for initiating the system as a whole.

The water supply source for the AFS is also redundant. The main source is by gravity feed from the condensate storage tank. This tank is sized to meet the normal operating and maintenance needs of the main turbine cycle systems; however, a minimum water level is maintained that is sufficient to remove residual heat generation for 24 hours at hot shutdown conditions.

An alternate supply of water to the pumps is provided by a connection to the 1.5 million gallon city water storage tank.

Ch.

11

The four original steam generators are stored completely intact, with all openings sealed with welded steel closure plates or bolted steel covers. The replaced primary elbow is also sealed at both ends with welded steel plates. On this basis, there will be no liquid or gaseous effluents released to the environment for the duration of storage of these components. The facility is designed to house the components until the entire plant is decommissioned.

11.1.2 System Design and Operation

The Waste Disposal System Flow Diagrams are shown in Figures 11.1-1A, B, 11.1-2A and 11.1-3. Typical Performance Data are given in Table 11.1-1. (The Indian Point 3 Environmental Tech. Specs., Part II, Section 5.3, require that a Semi-Annual Radioactive Effluent Release Report be submitted to the NRC. The data in Table 11.1-1 is from the report submitted to the NRC on February 15, 1991. As part of the review of the Authority's application to extend the IP3 operating license expiration date, the NRC reviewed IP3 solid waste shipments data for the period of 1986 to 1990. The NRC's conclusions were contained in an Environmental Assessment, issued by letter dated June 25, 1992.)

The Waste Disposal System was designed to collect and process all potentially radioactive primary plant wastes for removal from the plant site, within the limitations that were established by applicable governmental regulations. During system operation, fluid wastes are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary, before any attempt is made to discharge them and they are released under controlled conditions. A radiation monitor is provided to maintain surveillance over the release operation, but the permanent record of activity releases is provided by radiochemical analysis of known quantities of waste. The system was designed to process wastes generated during continuous operation of the primary system assuming that fission products, corresponding to defects in one percent of the fuel, escape into the reactor coolant. The liquid inventory of the plant is maintained within acceptable limits and releases are well below the limits of 10 CFR 20.

As secondary functions, system components supply hydrogen and nitrogen to primary system components as required during normal operation, and provide facilities to transfer fluids from inside the Containment to other systems outside the Containment.

11.1.2.1 System Description

Liquid Processing

The Waste Holdup System collects low level, radioactive liquid waste from throughout the facility and holds the waste until such time that it can be processed. The system consists of three tanks: the 24,500 gallon Waste Holdup Tank No. 31, which is located in the Waste Holdup Tank Pit, and the two 62,000 gallon Waste Holdup Tanks No. 32 and No. 33, which are located in the Liquid Radwaste Storage Facility. Waste Holdup Tanks No. 32 and No. 33 are connected in parallel to tank No. 31 and are provided with a pumped recirculation/spraying system to minimize precipitation of particulates and the accumulation of crud.

The Liquid Radwaste Storage Facility that houses Waste Holdup Tanks No. 32 and No. 33 is an underground concrete structure 75 feet long x 39'-6" wide x 24'-7" high. The 62,000 gallon tanks are supported on concrete piers. A sump pit is located in one of the corners of the building. To service the water tanks, and to interconnect the building with the Waste Holdup Tank Pit, a system of platforms is provided. In addition, an opening of 2'-6" x 7'-6" through the Waste Holdup Tank Pit wall forms the entrance from the Liquid Radwaste Storage Facility to the Waste Holdup Tank Pit. An emergency exit is provided by two openings in the roof of the structure, which is protected by a concrete penthouse. The two buildings are separated by a minimum 3-inch joint filled with seismic filler. The seismic joint adequately insures that in a seismic event both structures will react independently.

The building is supported on hard rock. The foundation consists of a rigid 2'-0" thick slab that is waterproofed. The waterproof membrane is laid upon a 4" concrete base. A 2-inch protection of concrete is placed over the waterproofing. The walls of the building are also waterproofed and they consist of reinforced concrete. The 3' thick reinforced concrete roof was poured on a steel deck and beam system.

The location of the opening in the Waste Holdup Tank Pit wall is such that it will not affect the structural integrity of the buildings. The east wall was designed to withstand a soil pressure of more than 24 feet. Locating the Liquid Radwaste Storage building adjacent to the Waste Holdup Tank Pit wall removes all of the earth pressure and the resultant stresses. The additional stresses imposed by the penetration are less than those that were imposed by the original loading condition. Therefore the net result is a safer condition of stress in the east wall.

To add operational flexibility in the event that the holdup capacity of the liquid WDS is exceeded, water from the holdup tank can be pumped to a demineralization system. This system consists of a series of pressure vessels containing activated charcoal and anion, cation and macro-reticular resins, and a pump to deliver water to the monitor tanks of the Chemical and Volume Control System. In addition, the Waste Holdup Tank pits are provided with a submersible pump tied to the inlet to waste tank No. 31.

During normal plant operation the Waste Disposal System processes liquids from the following sources:

a) Equipment drains and leaks

b) Radioactive chemical laboratory drains

c) Decontamination drains

d) Demineralizer regeneration

e) Floor drains

The system also collects and transfers liquid drained from the following sources directly to the Chemical and Volume Control System for processing:

a) Reactor coolant loops

b) Pressurizer relief tank

c) Reactor coolant pump secondary seals

d) Excess letdown during startup

e) Accumulators

f) Valve and reactor vessel flange leakoffs

The valve and reactor flange leakoff liquids flow to the Reactor Coolant Drain Tank. The reactor coolant drain tank water can drain directly tot he containment sump or can be discharged directly to the CVCS holdup tanks by the reactor coolant drain pumps. Thee pumps also return water from the refueling canal and cavity to the Refueling Water Storage Tank. To minimize contamination of the RWST, RCDT, and RCDT Pumps resulting from refueling operations, a filter system has been provided for the refueling cavity return flow to the Refueling Water Storage Tank. (See Section 9.3)

Where plant layout permits, waste liquids drain to the waste holdup tanks by gravity flow. Other waste liquids including floor drains drain to the sump and/or sump tank and are discharged to the waste holdup tanks by pumps operated automatically by a level controller.

If the preliminary analysis by sampling indicates that the liquid is suitable for discharge, it can be pumped from the waste holdup tank to the monitor tanks of the Chemical and Volume Control System (FSAR Section 9.2). When one monitor tank is filled it is isolated, and the waste liquid is recirculated and sampled for radioactive and chemical analysis while the second tank is in service. If analysis confirms that the contents are suitable for discharge, the waste liquid contained in the monitor tank is pumped to the service water discharge; otherwise, it is returned to the waste holdup tanks for reprocessing.

Although the radiochemical analysis forms the basis for recording activity releases, the radiation monitor provides surveillance over the operation by preventing the discharge valve from opening if the liquid activity level exceeds that which can be safely discharged.

Liquids in the holdup tanks not suitable for discharge are processed through the Liquid Radwaste Processing System skid.

Sampling of the condenser inlet water and discharge water system is done continuously.

Hudson River water samples are collected continuously from the intake structure (control location) and the discharge canal (indicator location), both of which are located on site. The sampling apparatus draws water from the intake structure and from the discharge canal and pumps it into respective containers. Each container has a volume that is approximately five gallons. One sample of inlet water and one sample of discharge water are taken, at a frequency specified by the Radiological Effluent Control Program, from the containers. Each of these samples is approximately four liters (one gallon). These samples are composited for monthly gamma spectroscopy analysis (GSA) and for quarterly tritium analysis.

Page 11.1-6
Deleted 7/89

Relief valves are provided for tanks containing radioactive waste if the tank might be overpressurized by improper operation or component malfunction. Tanks containing wastes that contain oxygen and are normally of low activity concentrations are vented into the Primary Auxiliary Building exhaust system.

11.1.3 Design Evaluation

Although radioactive liquid and gaseous releases and solid waste shipments from Indian Point 3 have at times exceeded those that were estimated for design, operating experiences demonstrate that the Waste Disposal System for the facility provides for suitable control of radioactive materials to the environment. A description of Indian Point 3 releases is submitted periodically to the NRC in the Radioactive Effluent Release Report, as described in the Technical Specifications, in accordance with the requirements of US NRC Regulatory Guide 1.21. The Indian Point 3 releases, as evidenced by the aforementioned reports, are well within the limits of 10 CFR 20.

Liquid Wastes

Liquid Wastes are generated by plant maintenance and service operations, and consequently, the quantities and activity concentrations of influents to the Waste Disposal System were, at the time of design, expected values. System loads have been greater than anticipated due to weather leaks and rainwater seepage.

The tritium concentration in a composite sample taken from every batch discharged to the river is determined periodically and used to establish the quantity of tritium released during that period.

As required by 10 CFR Part 20, every reasonable effort was made in the design of the Indian Point 3 Waste Disposal System to maintain radiation exposures and releases of radioactive materials to unrestricted areas as far below the limits specified in 10 CFR 20 as practicable.

In order to ensure that the design objectives were realized, several provisions were made for the radioactive waste processing systems. Specific design features were included in the design of the reciprocating charging pumps to collect leakage from these pumps and return it to the CVCS. The pressurizer spray valves, which are modulating in the RCS, have a live loaded packing configuration installed mitigate the potential for valve stem leakage. These two specific features were intended to reduce the amount of primary coolant leakage, the processing load, and consequently, the amount of activity being released.

Consideration was also given to continued plant operation with the existence of primary to secondary leakage. Technical Specifications limit primary to secondary leak rate to 0.3 gpm per steam generator.

A manually operated intertie was provided from the Indian Point 3 steam generator blowdown to the Indian Point 1 to Secondary Boiler Blowdown Purification System (SBBPS). Processing the Indian Point 3 blowdown through the Indian Point 1 SBBPS reduces releases by at least a factor of 10.

The requirements of 10 CFR 20 were satisfied in the design of the Indian Point Unit No. 3 Liquid Treatment System. Actual releases are reported semi-annually.

Gaseous Wastes

Gaseous Wastes consist primarily of hydrogen stripped from coolant discharged to the CVCS holdup tanks during dilution, nitrogen and hydrogen gases purged from the CVCS volume control when degasing the reactor coolant and nitrogen from the closed gas blanketing system. The gas decay capacity permits 45 days of decay for waste gas before discharge.

In the event of a pipe or tank rupture, the maximum anticipated quantity of waste gas that could be released from any one tank in the system is less than 50,000 curies of equivalent Xe-133, which would result in a dose of less than 0.5 rem beyond the site exclusion boundary.

Gaseous activity release to the plant vent on the primary Auxiliary Building (PAB) derives from reactor coolant leakage from various system components and from periodic discharges from the gas decay tanks in the Waste Disposal System.

As part of the 10 CFR 20 compliance analysis at the time of initial license application, reactor coolant leakage into the PAB was assumed to be 20 gallons per day at ambient temperature. The iodine release due to reactor coolant leakage into the PAB assumed a partition factor of 10^3 and an iodine removal efficiency of 99% for the charcoal filters in the Primary Auxiliary Building Ventilation System.

Gaseous activity releases from the Turbine Building were calculated for concurrent fuel defects and steam generator tube leakage. The turbine building gaseous releases were based on plant operation with an equivalent fuel defect percentage of 0.2% coincident with a steam generator to tube leakage of 20 gpd.

Nobel gas activities are released from the Turbine Building, primarily through the main condenser air ejector. Gaseous iodine is released from the main condenser air ejector. Gaseous iodine activity is also released from the Turbine Building as a result of steam and liquid leakage from various secondary system components into the building and exhaust from the gland seal condenser.

A system measuring the total effluent flow from the steam jet ejectors was installed in agreement with Regulatory Guide 1.97 in order to quantify the amount of radiation released through this path. The system consists of a sensor probe, mounted in the common exhaust from the after-condensers, and a remotely located electronic transmitter. The flow, ranging from 0 to 100 SCFM, may be monitored at the Critical Function Monitoring System (CFMS) upon demand. (The CFMS is described in Section 7.5)

Release from the main condenser air ejector were based on a maximum discharge rate of 60 SCFM and an iodine decontamination factor of 100 in both the steam generator and the main condenser. Releases due to steam and liquid leakage in the secondary system were based on leakages of 6 gpm of (cold equivalent condensed) steam and 12 gpm of (cold, condensed) liquid. An iodine decontamination factor of 100 in the steam generator was taken into account for both types of leakages and in addition a decontamination factor 3×10^3 for iodine was used for the liquid leakage. Releases from the gland seal condenser were based on an exhaust rate of 2.6 gal/min and an iodine decontamination factor of 100 in both the steam generator and the gland seal condenser. Actual releases were reported semi-annually.

Charcoal adsorbers were installed in the containment plant vent, the Primary Auxiliary Building, and the Fuel Handling Building ventilation systems.

Solid Wastes

Solid wastes consist of spent resins, spent filter cartridges, and miscellaneous contaminated materials such as paper, rag, and glassware. All solid radioactive wastes are packaged for removal to a license low-level waste burial facility in accordance with applicable regulations and burial site criteria. Waste volume and activities shipped offsite are reported semi-annually.

The Interim Radwaste Storage Facility (IRSF) may be utilized for temporary onsite storage of solid radioactive wastes and other non-liquid radioactive material. The facility design includes adequate shielding so that for up to and including a fully loaded facility, the doses to personnel outside the IRSF structure but within the IRSF protection fence shall be limited to less than 500 mrem/year and the offsite dose at the site boundary to less than 5 mrem/year. Two area radiation monitors with local annunciators are located in the facility.

All waste containers to be stored in this facility will be designed with materials compatible with the solid waste forms to prevent significant container corrosion. They are also designed to reduce the occurrence of uncontrolled releases of radioactive materials due to handling, transporting or storage.

Only solid radioactive waste will be stored in the facility. Floor drains are provided to collect any non-radioactive spilled liquid and are routed to a concrete sump tank located outside the facility. Before the tank is emptied, it will be sampled to ensure that the fluid is non-radioactive.

Fire protection is accomplished through the use of non-combustible construction materials, local fire extinguishers, and availability to an 8-inch fire main. A fire detection system with local annunciators will also be provided. Since the facility employs the use of non-combustible construction materials and storage containers, a fire suppression is not required. In the highly unlikely event of contaminated materials causing a fire, the exhaust louvres will be closed automatically.

11.1.4 Minimum Operating Conditions

Minimum operating conditions for the Waste Disposal System are dictated by the Radiological Effluent Controls, the Process Control Program (PCP), and the TRM.

11.2 RADIATION PROTECTION

11.2.1 Design Bases

The General Design Criteria presented and discussed in this section are those which were in effect at the time when Indian Point 3 was designed and constructed. These general design criteria, which formed the bases for the Indian Point 3 design, were published by the Atomic Energy Commission in the Federal Register of July 11, 1967, and subsequently made a part of 10 CFR 50.

The Authority has completed a study of compliance with 10 CFR Parts 20 and 50 in accordance with some of the provisions of the Commission's evaluations of compliance of Indian Point 3 with the General Design Criteria presently established by the Nuclear Regulatory Commission (NCR) in 10 CFR 50 Appendix A, were submitted to NRC on August 11, 1980, and approved by the Commission on January 19, 1982. These results are presented in Section 1.13

Monitoring Radioactivity Releases

Criteria: Means shall be provided for monitoring the containment atmosphere and the facility effluent discharge paths for radioactivity released form normal conditions, from anticipated transients, and from accident conditions. An environmental monitoring program shall be maintained to confirm that radioactivity releases to the environs of the plant have not bee excessive (GDC 17 of 7/11/67).

The containment atmosphere, the plant vent, the administration building vent, the containment fan-cooler's service water discharge, the Waste Disposal System gas and liquid effluent, the condenser air ejectors, the component cooling loop liquid, the component cooling water heat exchanger Service Water discharge, the discharges from the condensate polisher waste collection tanks and the steam generator blowdown are monitored for radioactivity released during normal operations, from anticipated transients, and from accident conditions. The fuel Storage Building and waste areas have no functional air monitoring, however, the HVAC Systems for these two areas are routed to the plant vent, which is monitored.

All gaseous effluent from possible sources of accidental releases of radioactivity external to the Reactor Containment (E.G., the spent fuel pit and waste handling equipment) will be exhausted form the plant vent which is monitored. All accidental spills in the auxiliary building are collected in a drain tank. Any Waste Disposal System liquid effluent discharged to the condenser circulating water canal is monitored. Any accidental spills from the Liquid Radwaste Processing System skid are collected in the Fuel Storage Building Cask Washdown area which drains to a sump and is pumped to the Waste Disposal System. For the case of leakage from the Reactor Containment under accident conditions, the plant area radiation monitoring system supplemented by portable survey equipment to be kept in Health Physics office area provides adequate monitoring of accident releases. The details of the procedures and equipment to be used in the event of an accident are given in the Emergency plan.

The discharges from the 20 code safety valves and the 4 power relief valves in the sTeam and Power Conversion System are not monitored by the radiation monitoring system, but the activity can be estimated from plant sampling, as the mass of the steam discharged can be determined and the activity concentration in the secondary side is known from periodic sampling.

Monitoring Fuel and Waste Storage

Criterion: Monitoring and alarm instrumentation shall be provided for fuel and waste storage and associated handling areas for conditions that might result in loss of capability to remove decay heat and to detect excessive radiation levels (GC+DC 18 of 7/11/67).

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect inadequate cooling and to detect excessive radiation levels.

Radiation monitors are provided to maintain surveillance over the waste release operation. The permanent record of activity releases I provided by radiochemical analysis of know quantities of waste.

There is a controlled ventilation system for the fuel storage and waste treatment areas of the auxiliary building which discharges to the atmosphere via the plant vent. Radiation monitors are in continuous service in these areas to actuate a high-activity alarm on the control board annunciator, as described in Section 11.2.3.

Fuel and Waste Storage Radiation Shielding

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities (GDC 68 of 6/11/67).

Adequate shielding for radiation protection is ensured during reactor refueling by conducting all spent fuel transfer and storage operations under water. This permits visual control of the operation at all times while maintaining low radiation levels for periodic occupancy of the area by operating personnel. The average exposure with 0.2% failed fuel that personnel could receive from the refueling water during fuel handling operations is 0.5mr/hr. The exposure to the crane operator moving an average fuel assembly is 3.4mr/hr. these dose rates are based on the expected activity during normal refueling operations. Pit water level is indicated, and water removed from the pit must be pumped out since there are no gravity drains. Shielding is provided for waste handling and storage facilities to permit operation within requirements of 10 CFR 20.

Gamma radiation is continuously monitored in the auxiliary building. A high level signal is alarmed locally and annunciated in the Control Room.

Protection Against Radioactivity Release from Spent Fuel and Waste Sotrage

Criterion: Provisions shall be made in the design of the fuel and waste storage facilities such that no undue risk to the health and safety of the public could results from an accidental release of radioactivity (GDC 69 of 7/11/67).

All fuel and waste handling and storage facilities are contained, and their related equipment were designed so that accidental releases directly to the atmosphere are monitored and do not exceed the limits of 10 CFR 100; refer to Sections 11.1.2, 15.2.2, and 14.2.3. The components of the Waste Disposal System are not subjected to any high pressure (Table 11.1-3) or stresses and are of Class I seismic design.

In addition, the tanks have a design pressure greater than atmospheric pressure, and the piping and valves of the system were designed t the codes given in Table 11.1-2. Hence, the probability of rupture or failure of the system is low.

The reactor cavity, refueling canal and present fuel storage pit are reinforced concrete structures with a steam-welded stainless steel plate liner. These structures were designed to withstand the anticipated earthquake loadings as seismic Class 1 structures so that the liner prevents leakage even in the event that the reinforced concrete develops cracks.

11.1.2 Shielding

Design Basis

Radiation shielding was designed for operation at maximum rated thermal power and to limit the normal operation radiation levels at the site boundary below those levels allowed for continuous non-occupational exposure. The plant is capable of continued safe operation with 1% fuel element defects.

The shielding provided was designed to ensure that in the event of a hypothetical accident, the integrated offsite exposure due to the continued activity will be below the limits established in 10 CFR 100.

A design review of Indian Point 3 was conducted in accordance with NUREG-0578, to identify areas, components and access paths which may require occupancy during post-accident recovery operation. The results of the review have been reported to NRC and remedial actions are being taken to ensure that all vital areas and equipment requiring access under post accident conditions meet the following criteria:

1) Continuous Occupancy – less than or equal to 15 mr/hr
2) Infrequent Access – less than or equal to 5 rem whole body dose, considering the required occupancy for the duration of the accident

Typical Zone 0 areas are the turbine building and turbine plant service areas and the Control Room. Typical Zone I areas are the offices, auxiliary building work stations and corridors, and the outer surfaces of the Containment and auxiliary building. Zone II areas would include the surface of the refueling water at refueling and the operating deck of the Containment during reactor shutdown. Areas designated Zone III include the sampling room, reactor cavity area after shutdown, and Reactor Containment penetration areas, including ventilation, steam line and electrical penetrations. Typical Zone IV areas include areas within the auxiliary building such as charging pump areas, evaporation area, heat exchanger areas, and valve operator areas. Typical Zone V areas are within the regions adjacent to the Reactor Coolant System at power operation and the demineralizer and volume control tank spaces.

All high radiation areas are appropriately marked and isolated in accordance wjth 10- CFR 20 and other applicable applications

The shielding is divided into five categories according to function. These functions include the primary shielding, the secondary shielding, the accident shielding, the fuel handling shielding and the auxiliary shielding.

Primary Shield

The primary shield is designed to:

1) Reduce the neutron fluxes incident on the reactor vessel to limit the radiation induced increase in transition temperature

2) Attenuate the neutron flux sufficiently to prevent excessive activation of plant components

3) Limit the gamma fluxes in the reactor vessel and the primary concrete shield to avoid excessive temperature gradients or dehydration of the primary shield

4) Reduce the residual radiation from the core, reactor internals and reactor vessel to levels which will permit access to the region between the primary and secondary shields after plant shutdown

5) Reduce the contribution of radiation leaking to obtain optimum division of the shielding between the primary and secondary shields.

Secondary Shield

The main function of the secondary shielding is to attenuate the radiation originating in the reactor and the reactor coolant. The major source in the reactor coolant is the Nitrogen-16 activity (83 mc/cc maximum), which is produced by neutron activation of oxygen during passage of the coolant through the core. The secondary shield was designed to limit the full power dose rate outside the Containment Building to less than 0.75 mr/hr.

Accident Shield

The main purpose of the accident shield is to ensure safe radiation levels outside the Containment Building following a maximum credible accident.

Fuel Handling Shield

The fuel handling shield was designed to facilitate the removal and transfer of present fuel assemblies and control rod clusters from the reactor vessel to the spent fuel pit. It was designed to attenuate radiation from spent fuel, control clusters, and reactor vessel internals, and together with exclusion gates, to reduce exposures to less than 2.0 mr/hr at the refueling cavity water surface and less than 0.75 mr/hr in areas adjacent to the spent fuel pit.

Auxiliary Shielding

The function of the shielding is to protect personnel working near various system components in the Chemical and volume Control System, the Residual Heat Removal System, the Waste Disposal System and the Sampling System.

The shielding provided for the auxiliary building was designed to limit the dose rate to less than 0.75 mr/hr in normally occupied areas, and at or below 2.0 mr/hr in intermittently occupied areas.

Shielding Design

Primary Shield

The primary shield consists of the core baffle, water annuli, core barrel (all of which are within the reactor vessel), the reactor vessel wall, and a concrete structure surrounding the reactor vessel.

The primary shield immediately surrounding the reactor vessel consists of an annular reinforced concrete structure extending from the base of the Containment to an elevation of 69 feet. The lower portion of the shield is a minimum thickness of 6 feet of regular concrete ($p = 2.3$ g/cm^3) and is an integral part of the main structural concrete support for the reactor vessel. It extends upward to join the concrete cavity over the reactor. The reactor cavity, which is approximately rectangular in shape, extends upward to the operating floor with vertical walls 4 ft. thick, except in the area adjacent to fuel handling, where the thickness is increased to 6 ft.

The primary concrete shield is air cooled to prevent overheating and dehydration from the heat generated by radiation absorption in the concrete. Eight "windows" have been provided in the primary shield for insertion of the out-of-core nuclear instrumentation. Cooling for the primary shield concrete and the nuclear instrumentation is provided by 12,000 cfm cooling air.

The primary shield calculated neutron flues and design parameters are listed in Table 11.2-2.

Secondary Shield

The secondary shield surrounds the reactor coolant loops and the primary shield. it consists of the annular crane support wall, the operating floor, and the Reactor Containment structure. The containment structure also serves as the accident shield.

The lower portion of the secondary shield above grade consists of the 4 ft-6 in cylindrical portion of the Reactor Containment and a 3 ft concrete annular crane support wall surrounding the reactor coolant loops. The secondary shield was designed to attenuate the radiation levels in the primary loop compartment from a value of 25 rem/hr to a level of less than 0.75 mr/hr outside the Reactor Containment Building. Penetrations in the secondary shielding are protected by supplemental shields.

The secondary shield design parameters are listed in Table 11.2-3.

Accident Shield

The accident shield consists of the 4 ft-6 in reinforced concrete cylinder capped by a hemispherical reinforced concrete dome of a 3 ft-6 in thickness. This shielding includes supplemental shields in front of the containment penetrations.

The equipment access hatch is shielded by a 3 feet 6 inches thick concrete shadow shield and a 1 foot 6 inches concrete roof to reduce scattered dose levels in the event of loss of reactor coolant accident accompanied by a complete core meltdown.

The accident shield design parameters are listed in Table 11.2-4.

Fuel Handling Shield

The refueling cavity, flooded to elevation 93.7 feet during refueling operations, provides a temporary water shield above the components being withdrawn from the reactor vessel. The water height during refueling is approximately 24-1/2 feet above the reactor vessel flange. This height ensures that a minimum of 10 feet of water will be above the active fuel of a withdrawn fuel assembly. Under these conditions, the dose rate is less than 2.5 mr/hr at the water surface (Reference: NSE 00-3-039 SFPC). This presumes a minimum pool level elevation of 93.2 feet. The spent fuel pit has a nominal level of 93.7 feet, which is half-way between the minimum and maximum water level alarm setpoints.

The refueling canal is a passageway connected to the reactor cavity and extending to the inside surface of the Reactor Containment. The canal is formed by two concrete walls each 6 feet thick, which extends upward to the same height as the reactor cavity. During refueling the canal is flooded with borated water to the same height as the reactor cavity.

The spent fuel assemblies and control rod clusters are remotely removed from the Reactor Containment through the horizontal spent fuel transfer tube and placed in the spent fuel pit. concrete, 6 feet thick, shields the spent fuel transfer tube. This shielding was designed to protect personnel from radiation during the time a spent fuel assemble is passing through the main concrete support of the Reactor Containment and the transfer tube.

Radial shielding during fuel transfer is provided by the water and the concrete walls of the fuel transfer pit. An equivalent of 6 feet of regular concrete is provided to insure a calculated maximum dose value of 0.75 mr/hr in the areas adjacent to the spent fuel pit. Exclusion gates are also provided in the fuel transfer tube.

Spent fuel is stored in the spent fuel pit which is located adjacent to the Containment Building. Shielding for the spent fuel storage pit is provided by 6 feet thick concrete walls and the pit is flooded to a level such that the water height is grater than 13 feet above the spent fuel assemblies.

The refueling shield design parameters are listed in Table 11.2-5.

Auxiliary Shielding

The auxiliary shield consists of concrete walls around certain components and piping which process reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods. Periodic access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so that the compartments may be entered without having to shut down and, possibly, to decontaminate the adjacent system.

The shield material provided throughout the auxiliary building is regular concrete)p=2.3 g/cm^3). The principal auxiliary shielding design parameters are tabulated in table 11.2-6.

The design basis used for shield design to allow access to manual backup items (e.g. valves) was that the integrated dose to an operator, immediately after the accident will be less than 3 Rems. Shielding platforms with integrated dose to an operator, immediately after the accident will be less than 3 Rems. Shielding platforms with reach rods to such valves have bee provided. The shielding will result in dose rates which are not significantly greater than the background dose from the containment (approximately 500 mr for one month following the accident). Doses in the vicinity of equipment located within the Primary Auxiliary Building would be much less due to the shielding afforded by the concrete walls of the Primary Auxiliary Building.

Shielding for the residual heat removal pumps is designed to limit the 8-hour integrated does to 3 Rems during maintenance of one residual pump with the adjacent pump circulating containment sump water. This was accomplished by the provision of shield walls around the pumps and associated piping and reach rods on the valves which must be manually operated.

11.2.3 Radiation Monitoring System

The radiation Monitoring System provides radiation detection equipment to insure safe operation of the plant.

The system was designed to perform three basic functions:

1) Warn operating personnel of any radiation health hazard which develop.

2) Give early warning of a plant malfunction which might lead to a health hazard or plant damage.

3) Prevent inadvertent release of radioactivity to the environment.

instruments are located at selected point in and around the plant to detect, indicate and record the radiation levels. If the radiation level should rise above the setpoint established for that channel, an alarm is initiated. The Radiation Monitoring System operates in conjunction with regular and special surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning is thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed 10 CFR 20 limits.

The only components of this system which are located in the Containment are the detectors for certain area monitoring channels. Some of these would not be expected to operate following a major Loss-of-Coolant Accident and were not designed for this purpose. Components of all other area and process monitoring channels were designed for post-accident, as required.

The components of the original Radiation Monitoring System were designed according to the following environmental conditions:

1) Temperature – an ambient temperature range of 40 to 120°

2) Humidity – 0 to 95%*

NOTE: *Equipment located in the control room area or other areas in the plant with controlled environments may be specified for smaller temperature and humidity ranges because of the controlled environment provided by the heating and ventilating system.

3) Pressure – Components in the Primary Auxiliary Building and the Control Room were designed for normal pressure. Area monitoring system components inside the Containment were designed to withstand pressure.

4) Radiation – Process monitors are of a non-saturating design so that they "peg" full scale if exposed to radiation levels up to 100 times full-scale indication. Such process monitors are located in areas where the normal and post-accident background radiation levels will not affect their usefulness.

   Process Monitors R20, R-62 A, B, C, D and R-63 A, B do not have this non-saturating design since they utilize a Geiger-Mueller tube. The range of these monitors is listed in Table 11.2-7, and these ranges are sufficient for these monitors to perform their functions.

Some monitors of the Radiation Monitoring System are required to meet the requirements of Regulatory Guide 1.97. These monitors are identified in subsections 11.2-3.1 and 11.2.3.2, and meet or exceed the design conditions stated above.

The Radiation Monitoring System is divided into the following subsystems:

1) The Process Radiation Monitoring System monitors various fluid steams for indication of increasing radiation levels.

2) The Area Monitoring System monitors area radiation in various parts of the plant.

3) The Environmental Radiation Monitoring Program monitors radioactivity in the area surrounding the plant. This program is outlined in the Off-site Dose Calculation Manual (ODCM).

In the Radiation Monitoring System, some monitors are located to detect radioactivity in several sample streams fed to a common header. These monitors read gross activity of the streams. Should a monitor detect a high gross activity, the lines into the common header may be isolated using valves located upstream of the header, and each stream read individually. In order to assure that the sampling lines into the header do not become plugged, they are periodically inspected and tested. On certain lines of high importance (e.g., the steam generator blowdown line) a flow meter indicates any variation in flow rate that would be caused by a stoppage in one of the lines.

11.2.3.1 Process Radiation Monitoring System

This system consists of channels which monitor various fluid streams for indication of increasing radiation levels. The channels and the type of radioactivity monitored are listed in Table 11.2-7A and the measurement ranges are given in Table 11.2-7. The monitors are described below and are designed to detect the minimum concentrations of the isotopes of interest and, in monitoring gross activity, are designed to generate an alarm under abnormal conditions. Isotopic identification and concentrations are determined by grab sample analysis.

Condensate Polisher Overboard Monitor (R-61)

This channel monitors liquid radioactivity of discharges from the LTDS or HTDS Waste Collection. Automatic valve closure is initiated by this monitor to prevent further release after a high radiation level is indicated and alarmed. This liquid monitor is part of a skid mounted, microcomputer-controlled offline sampling system containing a microprocessor, gamma sensitive scintillation detector, valves, control station and a flow switch. The detector output is transmitted to the microprocessor which converts the detector signal into digital and analog outputs for display, generates alarms and communicates with the Control Room Radiation Monitoring Cabinet. Alarms are provided in the Control Room and locally in the Condensate Polisher. The alarm trip setpoint for this process radiation monitor is established in accordance with the Indian Point 3 "Offsite Dose Calculation Manual." The trip setpoint ensures that the offsite radioactive releases are kept within 10 CFR 20 limits. Manual operation of a reset switch is required to reopen the discharge valves.

Main Steam Monitors (R-62 A-D)

Four radiation detectors are externally mounted next to the main steam lines outside the containment and upstream of the safety valves. These channels monitor the noble gases released through the main steam line safety valves and atmospheric dump valves during normal and accident plant operation. Local indications/alarms are in the upper cable tunnel penetration areas ass well as the control room.

The detection channels for the R-62 radiation monitor are designed to meet the range requirements of NUREG-0737. The range of these channels ($7.66E^{-03}$ to $7.66E^{+02}$ μCi/cc of ODCM instantaneous release mix) complies with a NUREG-0737 required range of $1.00E^{-01}$ to 1.00^{+0} μCi/cc of Xe-133 dose equivalent radioactivity in that the Xe-133 equivalent range for monitor R-62 is $6.04E^{-02}$ to $6.04E^{+03}$ μCi/cc.

The R-62 channels meet the intent of the range requirement imposed by Regulatory Guide 1.97 (i.e. 10^1 μCi/cc to 10^3 μCi/cc) in that they provide accurate monitoring of any radioactive releases through the main steam lines for the maximum steam line activity concentrations following a postulated design basis steam generator tube rupture accident. The actual detection range of the monitor has a low limit that is below the low limit of the range requirement imposed by Regulatory Guide 1.97 and a high limit that is above the maximum concentration of noble gases expected in the main steam lines on a design basis steam generator tube rupture accident. This range ($7.66E^{-03}$ to $7.66E^{+02}$ μCi/cc of ODCM instantaneous release mix) is displayed on the RM-23L digital display at the RM-80 microprocessor and on the RM-23A controller for the radiation monitor. The scales for the analog output from the monitor to the analog alarm and indication assemblies, the recorders and the QSPDS are $1.00E^{-03}$ to $1.00E^{+03}$ μCi/cc which bound the actual detection range.

Each of the Geiger-Mueller tube detectors is mounted in a led shield to minimize the effect of background radiation. Each detector transmits its output signal to a common microprocessor which converts the detector outputs to digital and analog signals for display locally and communicates with a radiation monitor controller located in the control room. This controller has a digital display of radiation levels for the operator which provides output signals for strip chart recorders and the Qualified Safety Parameter Display System (QSPDS).

Gross Failed Fuel Detector (R-63A, R-63B)

The Gross Failed Fuel Detector (GFFD) is based on the principle of measuring gamma radiation from fission products in the primary coolant after having allowed decay of the seven second half-life N-16.

Delay time is obtained by the length of tubing from the core to the detector. Piping to the detector is connected to the hot leg of the reactor coolant loop (Figure 11.2-6). The fluid passes a sample cooler before it reaches the two redundant detectors. The fluid passes through a flow meter and flow controller before draining into the volume control tank. The proper delay time (about 60 seconds) to the detector can be adjusted by regulating the rate of water flow. Figure 11.2-7 shows the block diagram of the GFDD.

No reactor limitations are imposed based on operability of this detector. The recommended operator action in conjunction with the use of the gross failed fuel detector are as follows:

1) Log the gross failed fuel detector reading once per shift and report any unusual count rate increase to the shift manager.

2) Have chemistry samples taken if the concentration exceed 5 μCi/cc. This change is indicative of some possible fuel element failures occurring.

Operational requirements relating to the GFFD are included in the Technical Specifications. These monitors meet the requirements of Regulatory 1.97.

Whenever the Gross Failed Fuel Monitor is inoperable, the sampling frequency shall be increased to twice per day, five days per week. The maximum time between analyses shall be sixteen hours for the two samples taken on a given day and three days between daily analysis. This accelerated sampling frequency need only be performed until the Gross Failed Fuel Monitor is declared operable.

<u>Design Containment Equilibrium Activities</u>

During normal plant operations, Radiation Monitoring Systems Channels R-11 and R-12 provide continuous indications of the containment atmosphere gross air particulate activity and gross gaseous activity, respectively. Backup monitoring during purging is provided by Radiation Monitoring system Channels, R-14, plant vent gas monitor and R-27, plant vent wide range gas monitor. Prior to either containment purge or pressure relieving operations, containment air samples are obtained and analyzed for both particulate and gaseous activities. Table 11.2-8 lists the anticipated design equilibrium containment activities following a 16-hour operation of the containment recirculation filtration system at an iodine removal efficiency of 99%. The operating basis reactor coolant leakage into the containment of 14.4 gal/day and reactor operation with 0.2% equivalent fuel defects are assumed. Table 11.2-9 shows calculated containment activities after recirculation filtration for 16 hours at a conservative iodine removal efficiency of 90% and assuming 50 lb/day reactor coolant leak rate into the Containment and 1% fuel defects.

The tritium level in the reactor coolant is monitored weekly, not to exceed 10 days between analyses. Measures are taken to insure that during refueling tritium activity in the refueling water is less than μCi/cc.

With containment purge at an assumed rate of 10,000 cfm, the maximum concentration of tritium in the containment air was calculated to be less than 1/5 of DAC.

The basis for this concentration was determined from the assumption that the refueling water evaporation rate is 100 lb/hr, the Containment is purged for 2 hours at an assumed rate of 10,000 cfm prior to access, and the purge continues during the refueling operation at an assumed rate of 10,000 cfm. The containment purge isolation valves will be shut prior to going above cold shutdown to ensure closing against accident conditions.

During normal plant operation, grab samples from the containment and auxiliary building areas are analyzed for tritium as required in the Technical Specifications.

Monitoring of Radioactivity Discharges

During normal plant operation all liquids discharged from the nuclear steam supply systems of the plant are released via the Waste Disposal System. Prior to discharging from the plant, samples are taken from the monitor tanks for isotopic analysis. In addition, all liquids discharged from the Waste Disposal System, are monitored by the Waste Disposal System Liquid Effluent Monitor R-18. This monitor provides automatic closure of flow control valve RCV-019 to assure discharges of less than 10 CFR 20 limits.

Proper operation of the monitor is assured by utilization of its check source and by comparison of the monitor reading to the monitor tank sample analysis. This sample analysis is taken to establish activity in the liquid to be discharged prior to its release from the plant.

Monitoring for the occurrence of primary to secondary leakage is provided by both the Condenser Air Ejector Monitor R-15 and the Steam Generator Blowdown Sample Monitor R-19. Upon indication of leakage by either of these monitors, means have been provided to manually divert the blowdown from Indian Point 3 to the Indian Point 1 Secondary Boiler Blowdown Purification System (SBBPS). From the standpoint of rapid determination of the occurrence of primary to secondary leakage, the two monitors (R-15 and R-19) provide redundancy for this function.

Proper operation of R-19 is assured by utilization of its check source and by comparison of the monitor reading to a blowdown liquid sample analysis. During those periods of operation with primary to secondary leakage and utilization of the Indian Point 1 SBBPS, the blowdown liquid is also monitored by the radiation monitor provided for the Indian Point 1 SBBPS before release to the environment.

During those periods of plant operation with primary to secondary leakage, monitoring for the subsequent occurrence of radioactivity from such leakage in the Condensate Polishing Facility waste effluent is provided by radiation detector R-61. Automatic valve closure is initiated by this monitor to prevent further release after a high radiation level is indicated and alarmed. Readout and alarm are in the Control Room and a local alarm is provided in the Condensate Polisher.

The release rate of radioactive liquid effluents from the site must be such that the concentration of radionuclides from the circulating water discharge does not exceed the limtis specified in 10 CFR 20, Appendix B, for unrestricted area.

Waste Disposal Processes

The Waste Disposal System for Indian Point 3 is described in Section 11.1. Performance dta are given in Table 11.1-1.

The Indian Point 3 liquid releases include discharges from the Waste Disposal System, steam generator blowdown, and Steam and Power Conversion System liquid leakage.

A manually operated intertie is provided from the Indian Point 3 steam generator blowdown to the Indian Point 1 Secondary Boiler Blowdown Purification System (SBBPS).

The radio iodine releases from the Blowdown Tank Vent Line are estimated using partition factors from Regulatory Guide 1.42. The radio iodine release is assumed to be 5% of the radio iodine activity released from the Steam Generator Blowdown when directed to the Blowdown Flash Tank.

The Indian Point 3 gaseous releases include pressure relief operations, reactor coolant leakage in the Primary Auxiliary Building, discharges from the Waste Disposal System, steam generator blowdown, and secondary system releases from the main condenser air ejector, the gland seal condenser and Steam and Power Conversion System steam and liquid leakage. Offsite doses from gaseous tritium releases are negligible.

Plant equipment is used in conjunction with developed operating procedure to maintain surveillance of radioactive gaseous and liquid effluents produced during normal reactor operations and expected operational occurrences in an effort to maintain radioactive releases to unrestricted areas as low as practicable.

Purge and Vent During Normal Operation

During normal reactor operations a "closed containment" is maintained. Containment purging occurs only during cold shutdown conditions.

It may be necessary, however, to provide containment pressure relief during normal operation. The flow rate and time period associated with the pressure relief are much less than that associated with the normal purging operation. Review of plant operating data for Indian Point 3 indicates that venting of the Containment is necessary once every two days for approximately one and one-half hours.

Prior to either containment purge or pressure relieving operations, containment air is sampled and analyzed for both particulate and gaseous activities.

A high radiation signal from either the Containment Air Particulate Monitor or the Containment Radioactive Gas Monitor initiates automatic closure of the containment supply and exhaust duct valves and pressure relief line valves. Both these monitors would be in operation during containment purging and venting operations. The monitors are located a few feet from the containment wall in the fan house at an elevation of 54'-9".

The Containment Radioactive Gas Monitor and the Plant vent Gas Monitor would both detect the radiation levels that would result from a fuel handling accident inside the Containment. High radiation level for the Containment Radioactive Gas Monitor initiates automatic closure of the containment purge supply and exhaust duct valves and pressure relief line valves.

Refer to the Technical Specifications and the Radiological Effluent Controls Program for applicable requirements relating to operability and periodic testing of these monitors plus related limitations placed on containment venting operations.

11.2.4 Health Physics Program

The Indian Point health physics program, medical emergency program and emergency plan are described in the "Indian Point 3 Emergency Preparedness Program" and the Indian Point 3 "Radiation Protection Plan."

11.2.5 Liquid Waste Release

All liquid waste releases are assayed for radioactivity prior to release to assure compliance with the limits established in the Technical Specifications.

11.2.6 Tests and Inspections

Complete radiation surveys were made throughout the plant containment and auxiliary building during initial phases of plant startup. Survey data were taken and compared to design levels at power levels of 10%, 50% and 100%, at rated full power. Survey data were reviewed for conformance to design levels before increasing to the next power range.

The Off-Site Dose Calculation Manual (ODCM) specifies surveillance requirements for Technical Specification required radiation monitors. The Technical Specification required effluent monitors are tested with calibrated sources at the designated calibration frequency and are tested daily using a remotely operated check source to verify the instrument response.

11.2.7 Handling and Use of Sealed Special Nuclear, Source and By-Product Material

A. Tests for leakage and / or contamination shall be performed as follows:

1. Each sealed source, with a half-life greater than thirty days, shall be tested for leakage and / or contamination at intervals not to exceed six months (see 11.2.7.A.2 for testing of sealed sources that are stored and not being used).

   NOTE: Does not apply to startup sources subject to core flux, tritium, and material in gaseous form.

2. Sealed sources that are stored and not being used shall be tested for leakage prior to any use or transfer to another user unless they have been leak tested within six months prior to the date of use or transfer. In the absence of a certificate indicating that a test has been made within six months prior to the transfer, sealed sources shall not be put into use until tested.

3. Startup sources shall be leak tested prior to being subjected to core flux and following repair or maintenance to the source.

B. Sealed sources are exempt from 11.2.7.A when the source contains:

1. Less than or equal to 100 microcuries of beta and / or gamma emitting material, or

2. Less than or equal to 5 microcuries of alpha emitting material.

C. The leakage test shall be capable of detecting the presence of 0.005 microcurie of radioactive material on the test sample.

D. If the leakage test reveals the presence of 0.005 microcurie or more of removable contamination, the sealed source shall immediately be withdrawn from use and either decontaminated and repaired, or be disposed of in accordance with USNRC regulations.

PAGE 11.2-30
DELETED 7/88

Ch.

12

General Manager - Operations

The General Manager -Operations (Figure 12.1-6B) is directly responsible to the Plant Manager. and the responsibilities of this position include:

- directing and coordinating the operation of the plant by overseeing the Operations, Chemistry and Performance and Reliability group functions and activities;
- assuring that the plant operates in an efficient, safe manner within the bounds of the Technical Specifications and other regulatory requirements.

Site Planning and Outage Services Manager

The Site Planning and Outage Services Manager reports to the Plant Manager and is responsible for:

- preparation of schedules in support of routine site work activities, including non-outage corrective maintenance, preventive maintenance, and surveillance tests;
- planning/preparation for and coordination of plant outages, including forced outages and refueling outages.

Operations Manager

The Operations Manager is responsible to the General Manager-Operations and the responsibilities of this position include:

- assuring that the plant is operated in accordance with approved procedures by qualified personnel;
- assuring that maintenance requests are properly transmitted, thus assuring that plant equipment is in a state of high reliability and readiness;
- providing the liaison between the shift and plant staff organizations;
- assuring that plant operation is conducted in full compliance with the Technical Specifications and all other regulatory requirements.

Either the Operations Manager or the Assistant Operations Manager holds or a Senior Reactor Operator License (SRO).

Assistant Operations Manager

The Assistant Operations Manager is responsible to the Operations Manager, and the responsibilities of this position include:

- direction of the functional conduct of shift operations; and
- assurance that plant operation is conducted in full compliance with the Technical Specifications and all other regulatory requirements.

Either the Assistant Operations Manager or the Operations Manager is required to hold a current NRC Senior Reactor Operator License.

Manager-Operations Support

The Manager-Operations Support is responsible to the Operations Manager. Responsibilities of this position include providing technical and administrative support to the operations department. Activities include:

- ensuring that operations department documentation is maintained in accordance with procedures,
- ensuring that routine audits and evaluations of department programs are performed, and
- establishing the shift schedule for operating crews in accordance with Technical Specification staffing requirements.

The Manager – Operations Support meets the qualification requirements of ANSI-N18.1-1971.

The Shift Technical Advisor (STA) reports to the Manager-Operations Support and is responsible for providing engineering expertise and advice to the Shift Manager and Control Room Supervisor on matters involving operational and nuclear safety.

Shift Manager

The Shift Manager is responsible to the Assistant Operations Manager for the operation of the plant on his shift. On off-shifts, weekends, and holidays, the Shift Manager represents plant management unless the Assistant Operations Manager, Operations Manager, any of the General Managers, or the SEO is onsite. The Shift Manager is responsible for:

- operation of the plant, in accordance with requirements of the NRC and other regulatory agencies;
- assurance that all operations on his shift are performed in accordance with approved procedures and are in compliance with the limits of the Technical Specifications;
- originating maintenance requests, as problems may arise;
- administrative implementation of plant security on off-shifts;
- maintaining an NRC Senior Reactor Operator License;
- performing the review and analysis of plant transients;
- determining the circumstance, analyzing the cause, and determining that operation can proceed safely before the reactor is returned to power after a trip or an unscheduled or unexplained power reduction; and
- providing direction for returning the reactor to power following a trip or an unscheduled or unexplained power reduction.

In case of radiation or any other hazard which, in the opinion of the Shift Manager, requires plant shutdown, the Shift Manager can order the plant shut down.

Field Support Supervisor (FSS)

The FSS reports to the Shift Manager and is manned at the discretion of the Operations Manager. The FSS is required to maintain a current SRO license and is responsible to the Shift Manager for supervision and coordination of operational activities outside the Control Room according to Administrative Procedures. The FSS may serve as the SRO assigned to supervise fuel handling

12.1.2.3 Lines of Communication

Major communications on plant operation, availability, scheduling and maintenance generally will be between the SEO and the CNO. However, should consultation on performance, anomalies or modifications be required, the SEO has at his disposal, for direct communications, the entire headquarters staff. Likewise, any of the supervisors at the plant having specific responsibilities may have direct communication with the engineer at headquarters assigned to that discipline or who is most cognizant of the area of concern.

Should the SEO be unavailable, the Plant Manager will assume his responsibilities or the SEO may delegate this responsibility to one of the General Managers or other qualified supervisory personnel.

Department Managers within the plant organization are responsible to the General Managers for the performance of specific duties. The General Managers report directly to the Plant Manager. The Plant Manager is responsible to the SEO for the functional performance of the plant. A Shift Manager, competent to supervise all shift operations, is on duty at all times, and has the authority to control all operating, maintenance and testing on his shift. At all times, the Shift Manager or Control Room Supervisor on duty has direct authority to shut down the plant if, in his opinion, it is required because of radiation or any other hazard.

Administrative Procedures originate from the SEO or his authorized representative. Safety related procedures are reviewed and approved in accordance with FSAR requirements.

12.1.2.4 Operating Shift Crews

The minimum requirements for shift crew composition, established in 10 CFR 50.54 (m)(2) and Section 5.2.2 of the Technical Specifications, are implemented by administrative procedures.

12.2.3 Qualification of Nuclear Plant Personnel

12.1.3.1 Qualification Requirements

The minimum qualifications with regard to educational background and experience for plant staff positions will meet or exceed the minimum qualifications of ANSI 18.1-1971 for comparable positions except for: (1) the Radiological and Environmental Services Manager who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975, and (2) the Shift Technical Advisor who shall have a bachelor's degree or equivalent in a scientific or engineering discipline with specific training in plant design and response and analysis of the plant for transients and accidents; and (3) the Operation Manager who shall meet

or exceed the minimum qualifications of ANSI N18.1-1971 except for the Senior Reactor Operator License requirement which shall be in accordance with the plant Technical Specifications.

Ch.

14

Reactor Trip

A reactor trip signal acts to open the two series trip breakers feeding power to the control rod drive mechanisms. The loss of power to the mechanism coils causes the mechanism to release the control rods, which then fall by gravity into the core. There are various instrumentation delays associated with each tripping function, including delays in signal actuation, in opening the trip beakers, and in the release of the rods by the mechanisms. The total delay to trip is defined as the time delay from the time that trip conditions are reached to the time the rods are free and begin to fall. The time delay assumed for each tripping function is as follows:

Tripping Function	Time Delay Seconds	Limiting Trip Point Assumed for Analysis
Overpower(nuclear),High Setting	0.5	118%
Overpower(nuclear) Low Setting	0.5	35%
Overtemperature ΔT	2.0	See Figure 7.2-11
Overpower ΔT	2.0	See Figure 7.2-11
High pressurizer pressure	2.0	2470 psia
Low pressurizer pressure	2.0	1750 psia
High pressurizer water level	1.5	100% of pressurizer level span
Low reactor coolant flow (from loop flow detectors)	1.0	87% loop flow
Undervoltage trip	1.5	Not Applicable*
Turbine trip	4.0	Not applicable
Low-low steam generator level	2.0	0% of narrow range level span
Under frequency	0.6	55Hz

The difference between the limiting trip pointer assumed for the analysis and the nominal trip point represents an allowance for instrumentation channel error and set point error. During preliminary startup tests, it was demonstrated that actual instrument errors are equal to or less than above assumed values.

*Analysis assumes Undervoltage Trip occurs at time zero, hence, no limiting trip point is explicitly modeled. Former limiting trip point assumed for analysis was 68% of nominal value.

Calorimetric Instrumentation Accuracy

The calorimetric error is the error assumed in the determination of core thermal power as obtained from the secondary plant measurements. The total ion chamber current (sum of the top and bottom sections) is calibrated (set equal) to this measured power on a periodic basis. The secondary power is obtained from measurement of feedwater flow, feedwater inlet temperature to the steam generator and steam pressure.

Reference 30 provides an equivalent uncertainty in rated power of 1.2% if the Leading Edge Flow Meter (LEFM) instrumentation is used.

Reference 30 also provides an equivalent uncertainty in rated power of 1.97% if the feedwater venturis are employed.

RCS Voiding During Transients

The voids generated in the reactor coolant system during anticipated transients are accounted for in the Westinghouse analysis models. Furthermore, based on transient analyses performed by Westinghouse using these models, it is concluded that steam voiding will not result in unacceptable consequences during anticipated transients. [4]

Reactor Coolant Pump Trip

The Safety Evaluation Report approved the Westinghouse Owners Group (WOG) methodology for justifying manual RCP trip in lieu of automatic trip. The three alternative trip criteria employed by WOG are consistent with the original RCP trip guidelines. Reactor Coolant Subcooling has been selected at Indian Point Unit 3, as the alternate criteria for determining when to trip the Reactor Coolant Pumps. The IP3 plant-specific information regarding the Reactor Coolant pump trip was reviewed by the USNRC[8]. Subsequently the safety evaluation report was issued by the USNRC[9] which determined that NYPA had satisfactorily addressed all of the points as identified in Generic Letter No. 85-12.

Plant Operating – Conditions

The analysis of the potential consequences of the inadvertent boron dilution event includes the following conservative assumptions:

- The analysis is performed for an inadvertent dilution of the RCS for power operation, startup, and refueling modes of plant operation.
- Conservative dilution flowrates have been assumed for each plant operating mode as already discussed. The effective dilution mass flowrate used in the analysis is greater than the nominal volumetric flowrate accounting for the differences in the densities of the primary coolant and the dilution source.
- During power operation (Mode 1), the initial boron concentration is assumed to be 1800 ppm which is a conservative maximum value for the conditions of hot full power, rods at the insertion limits and no xenon. The minimum reactivity change following a reactor trip, results in the maximum critical concentration for the conditions of hot zero power, all rods inserted except the most reactive RCCA, and no xenon. This minimum reactivity change is equivalent to 350 ppm. The critical concentration at hot-zero-power conditions is thus 1450 ppm.
- During Startup (Mode 2), the initial boron concentration is assumed to be 1800 ppm which is a conservative maximum value for the conditions of hot zero power, rods at the insertion limits and no xenon. The minimum reactivity change following a reactor trip, results in the maximum critical concentration for the conditions of hot zero power, all rods inserted except the most-reactive RCCA, and no xenon. This minimum reactivity change is equivalent to 250 ppm. The critical concentration at hot-zero-power conditions is thus 1550 ppm.
- During refueling (Mode 6), the initial boron concentration is assumed to be 1900 ppm which is a conservative minimum value which meets the refueling mode Core Operating Limits Report requirement for a shutdown margin of at least 5% $\Delta k/k$. The critical concentration is assumed to be 1330 ppm which is a conservative maximum predicted value for which the reactor will attain criticality during refueling conditions. The minimum change in boron concentration is thus 570 ppm.
- The dilution source is conservatively assumed to originate at 14.7 psia and 40°F
- The alarms alert the plant operator that a dilution is in progress.
- All other plant systems are assumed to be operating within the limits specified by the plant Technical Specifications and the Technical Requirements Manual.

Cases Considered

Four cases of the inadvertent boron dilution event are considered. The cases during power operation (Mode 1, manual and automatic rod control), startup (Mode 2), as well as the refueling case (Mode 6) are discussed.

Event Duration

Following the initiation of the inadvertent dilution flow into the RCS, the event duration for the current licensing-basis analysis is less than 40 minutes. Within this time frame, the following events have been assumed:

Power Operation (Mode 1)

- Alarm alerting the operator that an unplanned dilution of the RCS is progressing
- Operator takes action to terminate the dilution flow
- Loss of plant shutdown margin (if no operator action is taken)

During power operation with the reactor in automatic rod control, the power and temperature increase from the boron dilution causes the insertion of the control rods and a decrease in the available shutdown margin. The rod insertion limit alarms (LOW and LOW-LOW settings) alert the operator at least 38 minutes prior to losing the required minimum shutdown margin. This is sufficient time for the operator to determine the cause of dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

With the reactor in manual rod control and no operator action taken to terminate the transient, the power and temperature rise will cause the reactor to reach overtemperature ΔT trip setpoint resulting in a reactor trip. The boron dilution transient in this case is essentially the equivalent to an uncontrolled RCCA bank withdrawal at power. The maximum reactivity insertion rate for a boron dilution event is conservatively estimated about 2.0 pcm/sec, which is within the range of insertion rates analyzed for the uncontrolled RCCA bank withdrawal at power event. Thus, the effects of dilution prior to reactor trip are bounded by the uncontrolled RCCA bank withdrawal at power analysis as described in Section 14.1.2. Following reactor trip, there are greater than 15 minutes prior to criticality. This is sufficient time for the operator to determine the cause of the dilution, isolate the reactor water makeup source, and initiate boration before the available shutdown margin is lost.

Startup (Mode 2) and Refueling (Mode 6)

- Initiation of an unplanned dilution of the RCS
- Loss of plant shutdown margin (if no operator action is taken)

Following the termination of the dilution into the RCS, the operator can take action to initiate reboration and recover the lost shutdown margin.

Ch.

17

CHAPTER 17

QUALITY ASSURANCE PROGRAM

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLE OF CONTENTS

The Authority's Quality Assurance Program requires that purchase specifications contain or reference, as applicable: design information and technical requirements, including component and material identification requirements; including drawings, specifications, codes and industrial standards, regulatory guides and regulatory requirements; tests and inspection requirements; and special process instructions for such activities as fabrication, cleaning, erecting, packaging, handling, shipping, storing and inspecting. The specification contains, as appropriate, requirements which identify the documents to be prepared, maintained, submitted, and made available to the purchasing agent for review and/or approval. These documents include drawings, specifications, procedures, inspection and test records, inspection and fabrication plans, personnel and procedure qualifications, materials, chemical and physical test results. The specifications contain applicable requirements for the retention, control and maintenance of records, and the purchasing agent's rights of access to the vendor's facilities and records for source inspection and audit.

Procurement documents for spare or replacement parts shall be subject to requirements at least equivalent to those used for the original procurement. The original procurement documents may be used as a basis for purchase of spare or replacement parts.

The specifications contain provisions for extending applicable requirements of the document to subcontractors and suppliers including purchaser's right of access to such subvendor's facilities and records.

17.2.4.2 Preparation, Review, Approval and Issue of Procurement Documents and Changes

The Authority's Quality Assurance Program requires that procedures be established that clearly delineate the sequence of actions to be taken in the preparation, review, approval, and control of procurement documents.

The Authority's Quality Assurance requirements for procurement document control requires that document review procedures be established. The reviews of procurement documents in accordance with these procedures are performed by knowledgeable Quality Assurance personnel who can determine if quality requirements are complete, correctly defined and incorporated, that the procured items can be properly inspected and controlled, and that acceptance criteria are adequately specified.

The review and approval of procurement documents are documented prior to release and are available for verification.

The Authority's purchase specifications, and changes thereto, are reviewed by the Authority's Engineering, Procurement and Contract Administration/NuclearGeneration/System Engineering/Purchasing Coordinator, as applicable, in accordance with the requirements contained in the Authority's Quality Assurance Program. The Authority's procurement inventory department reviews purchase specifications thereto, for quality requirements in accordance with the Quality Assurance Program. Purchase orders are processed following approval of the specification by the Authority's Engineering, Nuclear Generation/System Engineering/Purchasing Coordinator, as applicable.

Changes and/or revisions to procurement documents shall be subject to the same degree of control as utilized in the preparation of the original document.

17.2.4.3 Vendor QA Program Requirements

The Authority requires that delegated organizations have documented Quality Assurance Programs and that subcontractors have and implement documented Quality Assurance Programs for materials, equipment, and services to an extent consistent with their importance to safety and scope activities.

The operating organization will perform Procurement Document Control activities at the plant in accordance with approved written procedures which conform to the requirements of the Authority's Quality Assurance Program.

17.2.4.4 Authority Controls

The Authority reviews, comments and concurs with preliminary procurement documents, final bid documents and any changes thereto, to assure that the applicable requirements of Appendix B to 10 CFR 50 are included.

The Authority will review recommendations for contract award and will assure that any exceptions to the contract will not affect quality requirements.

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with the approved QA Program requirements.

17.2.5 Instructions, Procedures and Drawings

17.2.5.1 General Description

The Authority's Quality Assurance Program requires that documents such as instructions, procedures and/or drawings which prescribe quality affecting activities be controlled and that quality affecting activities be performed in accordance with these instructions, procedures, and/or drawings, as applicable.

The operating organization will control instructions, procedures and drawings at the plant in accordance with approved written procedures which conform to the requirements in the Authority's Quality Assurance Program. Quality affecting activities performed by the operating organization shall be accomplished in accordance with these instructions, procedures and/or drawings, as applicable.

The Authority and/or delegated organizations shall have established measures for the control and implementation of instructions, procedures, and drawings for quality-related activities applicable to the scope of their responsibilities. The described measures by a designated A-E/Consultant, if applicable, are delineated in the document referenced in Section 17.2.2, Paragraph 17.2.2.1.

The procedure review and approval process shall be controlled and implemented by administrative procedure(s). Specific requirements regarding the review and approval of programs and procedures are as follows:

- Each program and procedure required by Technical Specifications and other procedures that affect nuclear safety, and changes thereto, shall be reviewed by a minimum of two designated technical reviewers who are knowledgeable in the affected functional area.

- Designated technical reviewers shall meet of exceed the qualifications described in Section 4 of ANSI N18.1-1971 for applicable positions, with the exclusion of the positions identified in Sections 4.3.2 and 4.5. Individuals whose positions are described in Section 4.3.2 and 4.5 may qualify as designated technical reviewers provided they meet the qualifications described in other portions of Section 4.

- The designated technical reviewers shall determine the need for cross disciplinary reviews. Individuals performing cross disciplinary reviews shall meet the same qualifications required for designated technical reviewers.

- Each program and procedure required by Technical Specifications and other procedures that affect nuclear safety, and changes thereto, shall be reviewed from a safety perspective by a qualified safety reviewer. Safety and / or environmental impact evaluations, when required, shall be reviewed by PORC.

- Nuclear safety related procedures and procedure changes shall be reviewed and approved, prior to implementation, by the appropriate member(s) of management.

- All changes to the Process Control Program (PCP) and the Offsite Dose Calculation Manual (ODCM) shall be reviewed by PORC and accepted by the Vice President prior to implementation.

Each procedure required by Technical Specification 5.4 and 5.5.3 and changes thereto, shall be approved prior to implementation by the appropriate responsible member(s) of management, as specified above. They shall also be reviewed periodically as set forth in administrative procedures.

Temporary changes to the procedures may be made provided:

- The intent of the original procedure is not altered.

- The change is approved by two members of the plant staff, at least on of whom holds a Senior Reactor Operator's License.

- The change is documented, and reviewed and approved by the appropriate member(s) of plant management, as specified above within 14 days of implementation.

17.2.5.2 Acceptance Criteria

The Authority through planned and periodic surveillance and audits of delegated organizations as well as selected audits of their vendors and subcontractors assures that the instructions, procedures, drawings and checklists used on safety-related equipment are controlled and implemented to meet the requirements of applicable codes, standards, regulatory guides and the QA Program.

Activities affecting quality are defined in specifications, drawings, procedures and instructions and include criteria for the acceptance of specific activities. These instructions, procedures or drawings shall delineate the applicable requirements of codes, standards, regulatory requirements and regulatory guides, and shall specify acceptance criteria. Accomplishment of tasks shall be documented, and shall include appropriate information that acceptance criteria have been met when required.

17.2.5.3 Authority Controls

The Authority will control and implement instructions, procedures and drawings for quality-related activities as follows:

1) Review, comment and concur with engineering specifications, selected drawings, procedures and instructions prepared for the performance of a quality related activity.

2) Prepare procedures related to the activities of this section which identify individuals or groups responsible for these activities.

3) Prepare procedures for design changes, maintenance, modifications, procurement, refueling, fabrication, installation, inspection, in-service inspection and cleaning.

4) Establish format and procedure for the preparation, review and approval of the Authority's Quality Assurance Procedures.

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with QA Program requirements.

17.2.6 Document Control

17.2.6.1 General Description

The Authority's Quality Assurance Program requires that documents affecting the quality of safety-related structures, systems and equipment during the operation phase of the plant be controlled.

The program establishes control such that: obsolete or superseded documents shall not be inadvertently used; changes shall be approved by the same group or individuals having authority and responsibility for the initial issue, or by other qualified responsible organizations delegated by the Authority; a method of revision level verification shall be provided; approved changes shall be promptly distributed; and applicable documents shall be available prior to the start of the work for which they are required.

The operating organizations will control documents affecting the quality of safety-related structures, systems and components at the plant in accordance with approved written procedure which conform to the requirements of the Authority's Quality Assurance Program

17.2.6.2 <u>Review and Approval of Documents</u>

The Authority's Quality Assurance Program requires that procedures and instructions be prepared to control the preparation, review, concurrence or approval, change or revision, issuance and distribution of documents such as the following:

1) Quality Assurance manuals, operating procedures and instructions.

2) Design specifications and drawings.

3) Manufacturing, design, construction and installation drawings.

4) Manufacturing and inspection, test, and special process procedures and instructions.

5) Procurement documents.

6) Administrative Procedures.

7) Maintenance Procedures.

8) FSAR and related design criteria documents.

9) Design change requests.

10) Nonconformance reports.

11) Technical Requirements Manual.

Figure 17.2-5 itemizes the Authority's Quality Assurance Program and QA Procedures and their relationship to the 18 criteria of 10 CFR 50 Appendix B.

17.2.6.3 <u>Authority Controls</u>

The Authority will control the preparation, review, approval and distribution of Authority documents including changes thereto which prescribe activities affecting quality performed by Authority personnel and assure that the latest issue of such documents are used. The Authority will provide for distribution of correspondence which address quality affecting activities, to affected Authority personnel. The Authority will maintain a master list or equivalent of issued Authority documents affecting quality related activities such as procedures, instructions and drawings.

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with approved QA Program requirements.

17.2.7 Control of Purchased Material, Equipment and Services

17.2.7.1 General Description

The Authority's Quality Assurance Program establishes controls to assure that purchased safety-related material, equipment and services, whether purchased directly or through contractors and subcontractors, conform to the procurement document requirements. These measures include, as appropriate, provisions for source evaluation and selection of equipment vendors, objective evidence of quality furnished by contractors or subcontractors, inspection and audit at the source, and examination of products upon delivery.

Vendor selection and evaluation is based on qualifying data, such as the seller's QA Program and past performance data of similar items and vendor surveys, to determine the adequacy of the facilities and the effectiveness of the QA Program. In lieu of annual evaluations, ongoing evaluations will use receipt inspection data, industry information and audit survey reports to assess vendor performance.

Objective evidence of quality furnished by the contractors or subcontractors shall identify the purchased material (e.g., codes, standards, specifications met by the materials or equipment). The contractors or subcontractors also required to identify any procurement requirements which have not been met together with a description of those nonconformances dispositioned "accept as is" or "repair" as part of this objective evidence of quality.

Source inspection shall be required when the conformance of materials, parts and components to procurement documents cannot be verified upon receipt or the service contracted is of a nature requiring a witnessing or verification function.

Receipt inspection includes verification that the required documentation has been received and that the items conform to the procurement documents. Receipt inspection shall be performed in accordance with written procedures and instructions, and receiving activities shall be documented to assure that: material, component, or equipment is properly identified, including the receiving documentation; acceptance records are inspected with predetermined inspection instructions; inspection records or certificates of conformance or compliance are available at the plant; items accepted and released are identified as to their inspection status. Personnel performing receipt inspection required by the QA Program will be certified in accordance with requirements for ASNI N45.2.6-1978.

The operating organization will perform activities related to control of purchased material, equipment and services at the plant, in accordance with approved written procedures which conform to the requirements of the Authority's Quality Assurance Program.

17.2.7.2 Major Supplier Evaluations

The Authority's Quality Assurance Program provides for the evaluation of Quality Assurance Programs of major suppliers and delegated organizations. These evaluations assure that they are capable of providing equipment, material and services which meet the applicable regulatory guides, codes, industry standards and regulatory requirements. Audits will be performed to verify that the major suppliers and delegated organizations are satisfactorily implementing approved QA Program requirements.

The Authority's Quality Assurance Program requires that qualified personnel evaluate the supplier capability to provide acceptable quality services and products before the award of the procurement order or contract. QA and engineering personnel participate in the evaluation of those suppliers providing critical components. The results of supplier evaluations are documented and filed.

17.2.7.3 Source and Vendor Evaluations

Based upon complexity of purchased items and supplier performance history, source inspection or audits of vendors shall be performed as necessary to assure that the required quality of the purchased items is obtained. Surveillance of suppliers' fabrication, inspection, testing, and shipment of materials, equipment and components will be planned, performed and reported in accordance with written procedures which assure conformance to the purchase order requirements. Source inspections, surveillances or audits of supplier activities shall be performed by qualified personnel from quality assurance, engineering and/or operations as determined necessary during the procurement phase.

Suppliers' certificates of conformance or compliance are periodically evaluated by audits, source or receiving inspection activities, independent inspections, or tests to assure their validity. Results of these evaluations are documented in appropriate reports.

The Authority's Quality Assurance Program requires that effectiveness of the control of quality by suppliers be assessed at intervals consistent with their importance and the complexity of the purchased items.

The Authority's Quality Assurance Program requires that spare or replacement parts of safety-related structures, systems, and components are subject to controls at least equivalent to those used for the original equipment.

The Authority may delegate Control of purchased material, equipment and service activities for plant structures, systems and components, and procurement quality control and activities including plant receiving inspection to an A-E/Consultant in accordance with Section 17.2.2, Paragraph 17.2.2.1. A-E/Consultant responsibility may include the preparation of specifications, drawings, and requisitions for the purchase of materials, equipment and services by the Authority. Included in this activity will be the quality evaluation of those vendors recommended for procurement. A-E/Consultant may provide inspection, surveillance and audit service at vendor facilities for plant equipment and on established notification points of selected delegated organization items.

17.2.7.4 Authority Controls

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with approved QA Program requirements.

17.2.8 Identification and Control of Materials, Parts and Components

17.2.8.1 General Description

The Authority's Quality Assurance Program requires that all organizations performing safety-related activities establish procedures to provide identification and maintain control of materials, parts and components, including partially fabricated assemblies to prevent the use of defective, unapproved

or incorrect materials and equipment. The procedures, as applicable, shall provide for the unique identification of items by serial numbers, part number or other appropriate means. The identification shall be placed on the item whenever practical and shall not degrade the quality or function of the item, or on records directly and readily traceable to the item. Verification of identification shall be accomplished at appropriate stages throughout manufacturing, shipment, receipt and installation.

The program provides measures which assure the traceability of items to records, which will verify conformance of the materials, parts and components to specified requirements (e.g., chemical and physical properties, tests, inspections, etc.), shall be provided from initial receipt of materials to installation, use, testing, and throughout the life of the item during operation, modification and repair. For consumable items traceability requirements shall be met by documentation which indicates that only acceptable materials have been used.

The operating organization will identify and maintain control of materials, parts and components at the plant in accordance with approved written procedures which conform to the Authority's Quality Assurance Program.

17.2.8.2 Authority Controls

The Authority will perform planned and periodic audits of delegated organizations to verify the program implementation in accordance with approved QA Program requirements.

17.2.9 Control of Special Processes

12.2.9.1 General Description

The Authority's Quality Assurance Program requires that special processes be adequately controlled.

A special process is defined as a unique manufacturing, inspection or test process where the assessment of quality by direct inspection of the process or product is disadvantageous or impractical after the operation is complete. Processes of this nature require the application of effective controls on the process as described later in this Section. Special processes include, but are not limited to, welding, cadwelding, studwelding, heat treating, nondestructive examination, protective coating application, concrete placement, and chemical cleaning.

The Authority's program requires that all organizations performing special processes on safety-related equipment at the plant or at manufacturing plants do so in accordance with approved procedures under controlled conditions, and that procedures, equipment and personnel shall be qualified in accordance with applicable codes, standards and specifications; such qualification records for plant personnel shall be maintained at the plant and kept current. Special process procedures shall reference the applicable codes, standards or specifications and provide methods of documenting accomplished activities.

The operating organization will control special processes at the plant in accordance with approved written procedures which conform to the requirements of the Authority's Quality Assurance Program.

17.2.9.2 Authority Controls

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with approved QA Program requirements.

17.2.10 Inspection

17.2.10.1 General Description

The Authority's Quality Assurance Program defines the program requirements that are applicable to inspections performed on safety-related equipment throughout all phases of operations.

The Quality Assurance Program shall provide inspection during manufacturing, shipping, receiving, storage, handling, installation, testing, operations, repairs, maintenance and modifications, as applicable. Inspection requirements shall be translated into written procedures, instructions and/or checklists. These documents shall govern the conduct and the degree of inspection activity to ensure that the required quality is obtained and objective evidence of the inspections is available.

The Authority's Quality Assurance Program requires that design specifications, drawings, procedures or instructions shall include the necessary inspection requirements including acceptance criteria to provide assurance that items and work conform to the design requirements.

Where direct inspection is not practicable, control of processing, equipment and personnel shall be used to determine acceptability.

When sampling plans are used, they shall be based on recognized standard sampling plans.

Inspection procedures, instructions or plans shall be made available where the activity is to be performed prior to the start of work. When notification or hold points have been established, either contractually by purchase documents or internally by the fabricator, or at the plant by the operating organization, the inspection program of plan provides, and the process control procedure shall include provisions to ensure that work does not progress beyond the notification points until released by the designated authority. The method of inspection shall be consistent with the complexity and nature of the work performed, i.e., nondestructive examination (NDE).

Qualified inspectors shall be perform inspection using proper equipment that has been calibrated in compliance with the requirements of Section 17.2.12. Inspectors shall be qualified in accordance with appropriate criteria and applicable codes, standards and training programs. Inspector qualifications and certifications shall be kept current.

Routine in-process inspections required during maintenance and operations shall be performed by qualified personnel of the plant staff of delegated organizations. Specific elements of work requiring quality acceptance shall be identified by the QA staff who shall perform the inspection and witness testing either at the plant or vendor facilities.

Acceptance inspection activities are performed by qualified inspection personnel who have not performed the work to be inspected. Inspection results are evaluated to determine that requirements have been satisfied.

Inspection requirements, based on applicable requirements of codes, standards and regulatory guides, are translated into inspection programs or plans by the manufacturing activities, to provide documented records of the inspection efforts required to assure quality. Inspection plans shall make use of in-process and final inspection operations, as required.

Maintenance and modification procedures are reviewed by qualified personnel to determine the necessary requirements related to such items as inspection, designation of inspection personnel and the need for documenting the results of subsequent inspections. Inspection of a repair or modification shall be performed by the same method and the same criteria as the original inspection, or by an approved alternate method.

Inspection operations, including monitoring, witnessing and/or auditing shall be documented and validated by inspection stamps and/or inspectors' sign-off.

17.2.10.2 Authority Controls

The Authority will perform planned and periodic audits of delegated organizations and participate in inspection at selected vendors' facilities, to verify program implementation in accordance with approved QA Program requirements.

17.2.11 Test Control

17.2.11.1 General Description

The Authority's Quality Assurance Program defines the basic requirements for all organizations performing tests on safety-related materials, equipment, components, systems and structures throughout all phases of operation.

Tests performed after modification, repair replacement shall be in accordance with the original design testing requirements or acceptable alternatives. The extent of testing shall be based on the complexity of the modifications, replacements or repairs. Acceptable alternatives must be approved by an appropriately designated organization.

The Authority's program requires that tests necessary to demonstrate that materials, equipment, components, systems and structures will perform satisfactorily in service shall be undertaken in accordance with written procedures, as required. These procedures are based on codes, standards, and applicable regulatory requirements. The test procedures to the extent applicable, shall include provisions to assure that all prerequisites have been met prior to further processing, such as the availability of appropriate calibrated equipment, completeness of the item, condition of the item, proper environmental conditions and arrangements for witness of mandatory tests by the Authority, contractor or authorized inspector. Test procedures shall be in sufficient detail and include caution or safety notes, so that the test operator's interpretation is not required.

Test personnel will be trained, qualified and certified, as necessary, for the various test functions. Test results shall be documented with sufficient detail to prevent misinterpretation. The organization that develops the design objectives or test limits, or other duly authorized organization, establishes acceptance criteria. Test results will be evaluated in accordance with the established criteria by a qualified, responsible individual or group. Test records will be filed and stored in an appropriate manner upon completion of the test and evaluation.

The operating organization will conduct tests and test control activities at the plant in accordance with approved written procedures which conform to the requirements of the Authority's Quality Assurance Program.

A summary report of appropriate plant testing shall be submitted to the NRC following (1), an amendment to the license involving a planned increase in power level, (2) installation of fuel that has a different design and (3) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the plant. The report shall address each of the tests identified in the FSAR and shall in general include a description of the measured values of the operating conditions or characteristics obtained during the testing and comparison of these values with acceptance criteria. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the Startup Report does not cover all three events (i.e., initial criticality, completion of startup programs, and resumption or commencement of commercial power operation), supplementary reports shall be submitted at least every three months until all three events have been completed.

17.2.11.2 Authority Controls

The Authority will review, comment and concur with selected written test procedures and specifications related to: testing, instrumentation and its maintenance and calibration, environmental conditions required for the performance of the test, and the acceptance limits relating to the test.

The Authority will review, comment and concur with the tests specified in procurement documents.

The Authority will review and comment on methods of documenting and recording test data and results.

The Authority will witness selected tests at vendor facilities and witness selected tests at the plant.

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with approved QA Program requirements.

17.2.12 Control of Measuring and Test Equipment

17.2.12.1 General Description

The Authority's Quality Assurance Program defines the requirements for the control of measuring and test equipment throughout all phases of measurement, inspection and monitoring of safety-related materials, components, systems and structures during operations. The Authority's program requires that all organizations performing measurement, inspection and testing of safety-related materials, components, systems, structures, and installation have written procedures must include calibration of all measuring and test equipment. The procedure must include calibration techniques,

calibration frequency, and reference transfer standards that are required. These procedures are for measuring and test equipment used in checking, repairing, maintaining, modifying or installing safety-related systems or components. It also includes permanent plant instruments whose calibration is a prerequisite of system surveillance and testing as required by the Technical Specifications.

All measuring and test equipment shall be uniquely identified and have traceability to the calibration records and technical data. Identification shall include the use of labels, tags, decals, etc., affixed to the equipment, when practical, denoting the date of calibration and the due date of next calibration.

Calibration frequency shall be dependent on the required accuracy, purpose, degree of usage, stability characteristics, manufacturer's recommendations, or other conditions affecting the measurement.

The reference and transfer standards shall be traceable to nationally recognized standards and, for any exceptions, provisions shall be made to document the basis for calibration. Calibration standards shall have an uncertainty (error) requirement of not more than 0.25 of the tolerance of the equipment being calibrated. A greater uncertainty, previous inspection requirements will be repeated using calibrated equipment.

The operating organization shall control measuring and test equipment used at the plant in accordance with approved written procedures which conform the requirements of the Authority's Operation Quality Assurance Program.

17.2.12.2 Authority Controls

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with approved QA Program requirements.

17.2.13 Handling, Storage and Shipping

17.2.13.1 General Description

The Authority's Quality Assurance Program defines the requirements for handling, storage and shipping activities performed on all safety-related equipment. The Authority's Program requires that organizations performing handling, storage and shipping activities do so to written procedures, as appropriate. These procedures, based on applicable requirements of codes and standards, have provisions for handling, cleaning, preservation, storage, packaging and shipping of equipment, as required.

The written procedures include detailed requirements for cleaning, coating and specifying environmental conditions for handling, storage and shipping.

The written procedures described special handling and precaution required during unloading or storage locations. The procedures contain the inspection instructions necessary to verify conformance with established criteria using qualified personnel, as required.

When necessary, the procedures specify the inspection and the inspection frequency of items in storage to preclude damage, loss or deterioration from environments such as corrosive atmosphere, moisture and temperature.

The operating organization will control handling, storage and shipping activities at the plant in accordance with approved written procedures which conform to the requirements of the Authority's Quality Assurance Program.

17.2.13.2 Authority Controls

The Authority will review specifications, drawings, procedures and instructions which contain the requirements for handling , storage, shipping, cleaning, preservation and maintenance of material and equipment whether in storage or installed at the plant as a structure, systems or components.

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with approved QA Program requirements.

17.2.14 Inspection, Test and Operating Status

17.2.14 General Description

The Authority's Quality Assurance Program describes the requirements for the control of inspection, test and operational status of all safety-related material, equipment and structures.

The organizations that perform tests and inspections on safety-related items of nuclear power plants have established measures for identifying the test, inspection or operating status of these items. Such measures conform to applicable codes and standards are implemented by written procedures which describe the use of indicators, such as tags, markings, shop travelers, stamps, route cards or inspection checklists that identify the status of the items at a given time.

Only authorized personnel are permitted to apply or remove tags, markings or stamps to or from the equipment and/or documentation. Stamps, such as for welding, inspection or test are controlled and documented such that the individual using the stamp is readily and uniquely identified.

The program assures that inspections and test performed out of sequence are adequately documented and do not compromise system integrity.

The procedures provide for positive identification and control of nonconforming items in accordance with Section 17.2.15 to prevent their inadvertent use.

The program requires that when bypassing or waiver of a designated QA inspection, test or critical work operation is required, it shall be controlled by cognizant QA personnel.

The operating organization will maintain the inspection, test, and operational status of all safety-related material, equipment and structures at the plant in accordance with approved written procedures which conform to the Authority's Quality Assurance Program.

17.2.14.2 Authority Controls

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with approved QA Program requirements.

17.2.15 Nonconforming Materials, Parts or Components

17.2.15.1 General Description

The Authority's Quality Assurance Program describes the requirements for the disposition, handling and control of nonconforming material during operation. The Authority's system for control of nonconformances provides measures for nonconformances, observed by Authority personnel, to be documented, reviewed, dispositioned and transmitted to the responsible delegated organization for corrective action. The documentation shall fully identify the item, nonconforming characteristics, inspection requirements, the specific requirements violated, the disposition of the nonconformances and the appropriate approval of the disposition.

The Program provides for the identification of personnel or group(s) responsible for assigning dispositions to nonconforming items. Dispositions authorizing a change in requirements shall be made by the same personnel or group(s) responsible for establishing the original requirement or by another authorized organization.

The Authority's program also requires that nonconforming items be clearly identified by appropriate means (tags, labels, stickers, markings, etc.) and segregated, if practical, until the disposition instructions for the nonconforming item has been received. The disposition of the nonconformances identified as "use as is" or "repair' are prepared or reviewed for acceptability by cognizant technical personnel.

Measures have been established in the program to assure that nonconformance data related to work performed at Contractor/major vendor's facilities, relative to "use as is" or "repair" dispositions are reflected in the inspection records and forwarded to the plant to be retained as part of the plant records following a review for acceptability by quality assurance and/or the cognizant technical personnel.

Acceptability of rework or repair of materials, parts, components, system, and structures is verified by reinspecting and retesting the item as originally inspected and tested or by a method which is equivalent to the original inspection and testing method; and inspection, testing, rework, and repair procedures are documented.

The program requires identification, classification, resolution ad follow up of material nonconformances which are detected during the course of operation activities. Periodic reviews are made of material nonconformance reports by Authority Quality Assurance personnel. These reviews are performed and results are documented and reported to appropriate management.

The Authority's program provides measures to prevent inadvertent use or installation of safety-related materials, parts or components when determined to be in noncompliance with the requirements of applicable codes, standards, drawings, specifications and procurement documents.

The operating organization will control nonconforming items at the plant in accordance with approved written procedures which conform to the requirements of the Authority's Quality Assurance Program.

17.2.15.2 Authority Controls

The Authority will review and comment on selected deficiency reports generated by delegated organizations at vendor facilities.

The Authority will verify at the plant that rework and repair activities are accomplished in accordance with disposition instructions.

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with approved QA Program requirements.

17.2.16 Corrective Action

17.2.16.1 General Description

The Authority's Quality Assurance Program describes the requirements for a corrective action program and assures that conditions adverse to or affecting quality are promptly identified, reported and corrected. The Authority's Program provides for systematic analysis of deficiencies, including nonconformance reports, the determination of the need for corrective action and the reporting to an appropriate level of the management, the condition, cause, and corrective action taken.

The Authority's Program requires identification of the significant conditions adverse to or affecting quality and the need for corrective action to be documented. The circumstances creating or contributing to the adverse condition, the action necessary to correct the condition, and measures taken to preclude recurrence are determined and documented by the organization responsible for implementing the needed corrective action.

Follow-up action is taken the verify that specified corrective action has been properly implemented. Verification of proper implementation, or any action taken which is not considered acceptable, is documented and distributed to appropriate levels of management. This distribution includes management of the organization responsible for implementation of the specified corrective action. Reports of the conditions adverse to quality are formally issued to appropriate levels of management of affected organizations.

Records are maintained to substantiate that these corrective action measures have been properly implemented. This corrective action system is implemented through the use of approved written procedures.

If the specified disposition for corrective action affects design of structures, systems or equipment, a technical review shall be made by the organization that established the original design criteria, or by other qualified responsible organizations delegated by the Authority, to verify adequacy of the stated disposition.

A corrective action system will be implemented at the plant in accordance with approved written procedures which conform to the requirements of the Authority's Quality Assurance Program.

17.2.16.2 Significant Deficiencies

Significant deficiencies which are determined to be within the scope of 10 CFR 21, will be reported to the NRC Directorate of Inspection and Enforcement in accordance with NRC regulatory requirements.

17.2.16.3 Authority Controls

The Authority will review and concur with corrective action in conjunction with nonconformance documents.

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with approved QA Program requirements.

17.2.17 Quality Assurance Records

17.2.17.1 General Description

The Authority's Quality Assurance Program requires that quality records of safety-related items and activities shall be identified, reviewed, retained and retrievable. These requirements are imposed on all organizations performing safety-related functions during operation. The Quality Assurance Program describes the requirements for record storage facilities which shall be constructed, located and secured in such a manner as to prevent destruction of the records through fire, flooding, theft and deterioration by environmental conditions. Records generated during the design, procurement and construction phases, shall be maintained and stored in the same described manner.

The Authority's program requires that records generated during the operation phase, documenting evidence of quality of items and activities including such items as: operating logs; principal maintenance and modification activities; results of reviews, inspections, tests, audits; abnormal occurrences; monitoring of work performance and material analysis; the qualification of personnel, procedures, and equipment; and other documentation such as drawings, specifications, procurement documents, calibration procedures, calibration reports, design changes and nonconforming and corrective action reports.

Requirements for identification, transmittal, retention, maintenance and review of quality related records are indicated in specifications, quality programs and procedures. Documentary evidence of these activities shall be available at the plant prior to release of material or equipment for installation.

The Quality Assurance Program specifies the type of information and data to be compiled for the inspection records, such as: description of operation; evidence of completion or verification of manufacturing, inspection or test operation; inspection and test results; information concerning nonconformances; and acceptability of the item tested or inspected. The qualifications and training records of test and inspection personnel shall also be documented.

The operating organization will maintain the store records at the plant in accordance with approved written procedures which conform to the Authority's Quality Assurance Program. Specific record retention requirements are listed in Appendix 17.2F.

17.2.17.2 Authority Controls

The Authority maintains records generated by Authority personnel, both onsite and offsite, in accordance with the requirements of the Quality Assurance Program.

The Authority will perform planned and periodic audits of delegated organizations to verify implementation in accordance with approved QA Program requirements.

17.2.18 Audits

17.2.18.1 General Description

The Authority's Quality Assurance Program includes a comprehensive system of planned and periodic audits to be carried out by the Authority Quality Assurance organization to verify compliance with all aspects of the program. This audit system provides data for continuing evaluation of the effectiveness of the program. In addition, surveillance audits are conducted routinely on and unscheduled basis ongoing or day-to0day activities to verify satisfactory completion of the activity.

The Authority will perform planned and periodic audits of delegated organizations to verify program implementation in accordance with appropriate QA Program requirements. Specific program areas are subsequently audited, consistent with the project schedule or where quality concerns are noted, so that the total program is reaudited within a scheduled period of time. The required reviews, audits, and relate record keeping requirements are listed in Section 17.2, Appendix E.

Audits are performed in accordance with pre-established procedures, checklists, etc., and conducted by trained personnel not having direct responsibilities in the areas being audited.

The Authority's audit program requires audit results to be documented, reviewed by or with management responsible for the area audited, and appropriate action initiated to correct any deficiencies. The organization conducting the audit is responsible for conducting the follow-up actions including reaudit of deficient areas to assure correction of the discrepancies. Results of audits are summarized in audit reports which are reviewed by Quality Assurance.

The Authority's audit program, as defined in the Authority's Quality Assurance Program includes the following types of audits to provide a comprehensive, independent verification and evaluation of all quality related procedures and activities to assure they are in compliance with the Authority's established program requirements:

1) Audits of delegated organizations

2) Audits of selected vendors and contractors

3) Audits of plant operation activities

4) Audits internal to the Authority

Independently, or concurrent with the audits and inspections by delegated organizations, the Authority may conduct audits of vendors and contractors such as equipment fabricators, material supplier, consultants and various contractors working on plant activities.

17.2.18.2 Authority Controls

The Director-QA, based on his review, reports audit findings and the actions to be taken to correct the deficient conditions to the CNO. These reports also serve as a source of information for the Authority's Quality Assurance Program evaluation by management.

The Authority will perform planned and periodic audits at selected vendors' facilities to verify program implementation in accordance with approved QA Program requirements.

Pages 17.2-31 through 32
Deleted 7/95

Appendix E

Q.A Reviews and Audits

17.2.E.1 PLANT OPERATING REVIEW COMMITTEE (PORC)

17.2E.1.1 FUNCTION

The Plant Operating Review Committee shall function to advise the Vice President – Operations on all matters related to nuclear safety and all matters which could adversely change the plant's environmental impact.

17.2E.1.2 COMPOSTION

The Plant Operating Review Committee shall be composed of a Chairman, Vice Chairman and members designated in writing by the Vice President – Operations. Members shall collectively have responsibility in at least the areas of operations, maintenance, radiation safety, chemistry, engineering, instrumentation and controls, reactor engineering, nuclear licensing, and quality assurance.

17.2E.1.3 ALTERNATES

All alternate members shall be appointed in writing by the (PORC) Chairman to serve on a temporary basis.

17.2E.1.4 MEETING FREQUENCY

The PORC shall meet at least once per calendar month and as convened by the PORC Chairman or his designated alternate.

17.2E.1.5 QUORUM

A quorum of the PORC shall consist of the Chairman or Vice-Chairman, and at least five members including alternates. Vice-Chairman may act as members when not acting as Chairman. A quorum shall contain no more than two alternates.

17.2E.1.6 RESPONSIBILITIES

The Plant Operating Review Committee shall be responsible for:

a) Review of 10 CFR safety and environmental impact evaluation associated with procedures and programs required by Technical Specifications and changes thereto.

b) Review of all proposed tests and experiments that affect nuclear safety.

c) Review of all proposed changes to the Operating License and Technical Specifications.

d) Review of all proposed changes or modifications to plant systems or equipment that affects nuclear safety.

e) Review of changes to the Process Control Program and the Offsite Dose Calculation Manual.

f) Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the Vice President – Operations, who will forward the report to the Chief Operating Officer, Director – Licensing and the Chairman of the Safety Review Committee.

g) Review of all reportable events.

h) Review of facility operations to detect potential nuclear safety hazards.

i) Performance of special reviews, investigations or analyses and reports thereon as required by the Vice President – Operations or the Chairman of the Safety Review Committee (SRC).

j) Review of every unplanned onsite release of radioactive material to the environs including the preparation of reports covering evaluation, recommendations and disposition of the corrective action to prevent recurrence and the forwarding of these reports to the Vice President – Operations and to the Safety Review Committee.

17.2E.1.7 AUTHORITY

The Plant Operating Review Committee shall:

a) Recommend to the Vice President – Operations approval or disapproval of items considered under 17.2E.1.6, items (a) through (e).

b) Render determination with regard to whether or not each item considered under 17.2E.1.6, items (a) through (e) does not require prior NRC notification or a License Amendment as defined in 10 CFR 50.59.

c) Provide notification within 24 hours to the Chairman of the SRC and the Chief Operating Officer of disagreement between the PORC and the Vice President – Operations; however, the Vice President – Operations shall have responsibility for resolution of such disagreements.

17.2E.1.8 RECORDS

The plant Operating Review Committee shall maintain minutes of each meeting and copies shall be provided to the Chairman of the SRC and the Chief Operating Officer.

17.2E.2 SAFETY REVIEW COMMITTEE (SRC)

17.2E.2.1 FUNCTION

The SRC shall function to provide independent review and audit of designated activities in the areas of:

a) Nuclear power plant operations
b) Nuclear engineering
c) Chemistry and radiochemistry
d) Metallurgy
e) Instrumentation and control
f) Radiological safety
g) Mechanical engineering
h) Electrical engineering
i) Administrative controls and quality assurance practices
j) Environment
k) Civil / Structural Engineering
l) Emergency Planning
m) Nuclear Licensing
n) Other appropriate fields associated with the unique characteristics of the nuclear power plant.

17.2E.2.2 CHARTER

The conduct of the SRC will be in accordance with a charter approved by the Chief Operating Officer. The charter will define the SRC's authority and establish the mechanism for carrying out its responsibilities.

17.2E.2.3 MEMBERSHIP

The SRC shall be composed of at least six individuals including a Chairman and a Vice Chairman. Members shall be appointed by the Director – Licensing and approved by the Chief Operating Officer. SRC members and alternates shall have an academic degree in engineering or a physical science, or the equivalent, and shall have a minimum of five years technical experience in one or more areas listed in 17.2E.2.1.

17.2E.2.4 ALTERNATES

Alternates for the Chairman, Vice Chairman and members may be appointed in writing by the Director – Licensing and approved by the Chief Operating Officer.

17.2E.2.5 CONSULTANTS

Consultants may be used as determined by the SRC Chairman and as provided for in the charter.

17.2E.2.6 MEETING FREQUENCY

The SRC shall meet at least once per six months.

17.2E.2.7 QUORUM

A quorum shall consist of at least a majority of the appointed individuals (or their alternates) and the Chairman (or the designated alternate). No more than two alternates may participate as SRC voting members at any one time. No more than a minority of the quorum shall have direct line responsibility for the operation of the plant.

17.2E.2.8 REVIEW

The SRC shall review:

a. The Safety evaluations for 1) changes to procedures, equipment or systems and 2) tests or experiments completed under the provision of Section 50.59, 10 CFR, to verify that such actions did not constitute an unreviewed safety question.

b. Proposed changes to procedures, equipment or systems which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

c. Proposed tests or experiments which involve an unreviewed safety question as defined in Section 50.59, 10 CFR.

d. Proposed changes to Technical Specifications of this Operating License.

e. Violations of codes, regulations, orders, Technical Specifications, license requirements, or internal procedures or instructions having nuclear safety significance.

f. Significant operating abnormalities or deviations from normal and expected performance of plant equipment that affect nuclear safety.

g. ALL REPORTABLE EVENTS.

h. All recognized indication of an unanticipated deficiency in some aspect of design or operation of safety related structures, systems, or components.

i. Reports and meetings minutes of the Plant Operating Review Committee.

17.2E.2.9 AUDITS

The following audits shall be performed under the cognizance of the SRC.

a. The conformance of facility operation to provisions contained within the Technical Specifications and applicable license conditions at least once per 24 months.

b. The training and qualifications of the entire facility staff at least once per 24 months.

c. The results of actions taken to correct deficiencies occurring in facility equipment, structures, systems or methods of operation that affect nuclear safety at least once per 24 months.

d. The performance of activities required by the Operational Quality Assurance Program to meet the criteria of Appendix "B," 10 CFR 50, at least once per 24 months.

e. Any other area of facility operation considered appropriate by the SRC of the Chief Operating Officer.

f. The Facility Fire Protection Program and implementing procedures at least once per two years.

g. A fire protection and loss prevention inspection and audit shall be performed annually utilizing either qualified offsite licensee personnel or outside fire protection firm.

h. An inspection and audit of the fire protection and loss prevention program shall be performed by an outside qualified fire consultant at intervals no greater than 3 years.

i. The radiological environmental monitoring program and the results thereof least once per 24 months.

j. The OFFSITE DOSE CALCULATION MANUAL and implementing procedures at least once per 24 months.

k. The PROCESS CONTROL PROGRAM and implementing procedures for processing and packaging of radioactive wastes at least once per 24 months.

17.2E.2.10 <u>AUTHORITY</u>

The SRC shall advise the Chief Operating Officer on those areas of responsibility specified in 17.2E.2.8 and 17.2E.2.9.

17.2E.2.11 <u>RECORDS</u>

Records will be maintained in accordance with ANSI 18.7-1972. The following shall be prepared and distributed as indicated below:

a) Minutes of each SRC meeting shall be prepared and forwarded to the Chief Operating Officer within 30 days after the date of the meeting.

b) Reports of reviews encompassed by 17.2E.1 shall be prepared and forwarded to the Chief Operating Officer within 30 days following completion of the review.

c) Audit reports encompassed by 17.2E.2 shall be prepared and forwarded to the Chief Operating Officer and to the management positions responsible for the areas audited within 30 days after the completion of the audit.

d) Each REPORTABLE EVENT shall be reviewed by the PORC and a report submitted by the Vice President – Operations to the Chief Operating Officer, Director - Licensing and the Chairman of the SRC.

Appendix F

RECORD RETENTION

17.2F.1 The following records shall be retained for at least five years:

a) Records and logs of facility operation covering time interval at each power level.

b) Records and logs of principal maintenance activities, inspection, repair and replacement of principal items of equipment related to nuclear safety.

c) All REPORTABLE EVENTS submitted to the Commission.

d) Records of surveillance activities, inspections and calibrations required by the Technical Specifications.

e) Records of changes made to Operating Procedures.

f) Records of radioactive shipments.

g) Records of sealed source and fission detector leak tests and results.

h) Records of annual physical inventory of all source material of record.

i) Records of reactor tests and experiments.

17.2F.2 The following records shall be retained for the duration of the Facility Operating License:

a) Records of any drawing changes reflecting facility design modification made to systems and equipment described in the FSAR.

b) Records of new and irradiated fuel inventory, fuel transfers and assembly burnup histories.

c) Records of facility radiation and contamination surveys.

d) Records of radiation exposure as required by 10 CFR 20.

e) Records of gaseous and liquid radioactive material released to the environs.

f) Records of transient or operational cycles for those facility components designed for a limited number of transient cycles.

g) Records of training and qualifications for current members of the plant staff.

h) Records of in-service inspections performed pursuant to the Technical Specifications.

i) Records of Quality Assurance activities required by the QA manual.

j) Records of reviews performed for changes made to procedures or equipment or reviews of tests and experiments pursuant to 10 CFR 50.59.

k) Records of meetings of the PORC and SRC.

l) Records for Environmental Qualification which are covered under the provisions of paragraph 6.13.

m) Records of secondary water sampling and water quality.

n) Records of analyses required by the radiological environmental monitoring program that would permit evaluation of the accuracy of the analysis at a later date. This should include procedures effective at specified times and records showing that these procedures were followed.

o) Records of service lives of all safety-related hydraulic snubbers including the date at which the service life commences and associated installation and maintenance records.

p) Records of reviews performed for changes made to the Offsite Dose Calculation Manual (ODCM) and the Process Control Program (PCP).

Tables

TABLE 3.2-4
(Sheet 1 of 3)

THERMAL AND HYDRAULIC DESIGN PARAMETERS
(For Cycle 1 and Cycle 11)

	Cycle 1	**Cycle 11**
Total Heat Output, MWt	3025	Same as Cycle 1
Total Heat Output, Btu/hr	$10{,}324 \times 10^6$	Same as Cycle 1
Heat Generated in Fuel, %	97.4	Same as Cycle 1
Maximum Thermal Overpower, %	112	Same as Cycle 1
Nominal System Pressure, psia	2250	Same as Cycle 1
Hot Channel Factors		
Heat Flux		
Engineering, F_q^E	1.03	Same as Cycle 1
Total, F_q^{T*}	2.32	2.42
Enthalpy Rise – Nuclear $F_{\Delta H}^N$	1.55	(V+)1.70 (V5)1.65
Coolant Flow		
Total Flow Rate, lbm/hr	130.2×10^6	138.5×10^6
Average Velocity Along Fuel Rods, ft/sec	14.9	17.02
Average Mass Velocity, lbm/hr•ft^2	2.43×10^6	2.78×10^6

*The total heat flux hot channel factor is a generic limit. The actual value is documented in the COLR.

Table 7.5-1
Regulatory Guide 1.97 Instruments Required

REG GUIDE 1.97 STATUS OF COMPLIANCE

INDEX	TYPE CAT	VARIABLE ONE	VARIABLE TWO	INST LOOP	NOTES
106B	A1	Containment	Wide Range Pressure, Channel I	P1421	O
106C	A1	Containment	Wide Range Pressure, Channel II	P1422	O
107A	A1	Steam Generator 31	Pressure, Channel I	P419A	
107B	A1	Steam Generator 31	Pressure, Channel II	P419B	
107C	A1	Steam Generator 31	Pressure, Channel IV	P419C	
107D	A1	Steam Generator 32	Pressure, Channel I	P429A	
107E	A1	Steam Generator 32	Pressure, Channel II	P429B	
107F	A1	Steam Generator 32	Pressure, Channel IV	P429C	
107G	A1	Steam Generator 33	Pressure, Channel I	P439A	
107H	A1	Steam Generator 33	Pressure, Channel II	P439B	
107I	A1	Steam Generator 33	Pressure, Channel IV	P439C	
107J	A1	Steam Generator 34	Pressure, Channel I	P449A	
107K	A1	Steam Generator 34	Pressure, Channel II	P449B	
107L	A1	Steam Generator 34	Pressure, Channel IV	P449C	
108A	A1	Refueling Water Storage Tank	Level, Alarm	L920	N
108B	A1	Refueling Water Storage Tank	Level, Alarm	L921	N
109A	A1	Containment Water Level	Level	L1253	L
109B	A1	Containment Water Level	Level	L1254	L
111A	A1	Containment	Radiation, Area, High Range	R25	
111B	A1	Containment	Radiation, Area, High Range	R26	
112A	A1	Secondary Cooling	Radiation, Main Steam	R62	SS
113A	A1	Primary Coolant	Temperature, Core Exit	N/A	
114A	A1	Condensate Storage Tank Level	Level	L1128	
114B	A1	Condensate Storage Tank Level	Level	L1128A	
115A	A1	RCS Subcooling	Temperature		M
115B	A1	RCS Subcooling	Temperature		M
201A	B1	Neutron Flux Excore	Radiation, Intermediate Range Channel I	N38	
201B	B1	Neutron Flux Excore	Radiation, Intermediate Range Channel II	N39	
202A	B3	Control Rods	Position	N/A	

REG GUIDE 1.97 STATUS OF COMPLIANCE

INDEX	TYPE CAT	VARIABLE ONE	VARIABLE TWO	INST LOOP	NOTES
502M	E2	PAB 41'	Radiation, Area, Hall On Column Next To Containment Spray Pumps	N/A	X
502N	E2	PAB 34'	Radiation, Area, Hall Near Entry To Safety Injection Pumps	R66	
502P	E2	PAB 41'	Radiation, Area, Pipe Tunnel In Area Of Chemistry Post Accident Sampling Station	R67	
502Q	E2	PAB 15'	Radiation, Area, On North Wall Adjacent To RHR Valve Gallery	R68	
502R	E2	RAB 15'	Radiation, Area, Hall On Wall At Entry To Filter Cell	N/A	X
502S	E2	PAB 54'	Radiation, Area, Within The Doorway On The Wall, Pipe Penetration	R69	
502T	E2	PAB 67'	Radiation, Area, Above Pipe Penn In Area Of Hydrogen Recombiner Panels	N/A	X
502U	E2	Fan Building 92'	Radiation, Area, In Area Of 4 Channel Iodine Monitors	R70	
502V	E2	Fan Building 72'	Radiation, Area, Outside Plenum In Area Of Differential Pressure Instruments	R70	
503A	E2	Containment	Radiation, Effluent, Noble Gas		
504A	E2	Reactor Shield Building Annulus	Radiation, Effluent, Noble Gas		
505A	E2	Auxiliary Building	Radiation, Effluent, Noble Gas, Or Others Containing Primary System Gases		
506A	E2	Cond Air Removal Sys Exhaust	Radiation, Effluent, Noble Gas	R15	NN
506B	E2	Cond Air Removal Sys Exhaust	Radiation, Effluent, Noble Gas – Flow Rate	R15	
507 A	E2	Common Plant Vent	Radiation, Effluent, Noble Gas	R27	SS
507B	E2	Common Plant Vent	Radiation, Effluent, Flow Rate	R27	SS
508A	E2	Steam Generator	Radiation, Effluent, Noble Gas From Safety Relief Valves Or Atm Dump Valves	R62	FF
509A	E2	Admin Bldg Exhaust Vent	Radiation, Effluent, Noble Gas From 4th Floor	R46	OO, CC
509B	E2	Admin Bldg Exhaust Vent	Radiation, Effluent, Flow Rate, 4th Floor	NONE	OO
509C	E2	Radioactive Machine Shop Exhaust Vent	Radiation, Effluent, Noble Gas	R59	
509D	E2	Radioactive Machine Shop Exhaust Vent	Radiation, Effluent, Flow Rate	FT-1776	
509E	E2	Steam Generator Blowdown	Radiation, Effluent	R19	
509F	E2	Steam Generator Blowdown	Radiation, Effluent, Flow Rate	F538	
510A	E3	Common Plant Vent	Radiation, Effluent, Particulates	N/A	EE, SS
510B	E3	Common Plant Vent	Radiation,, Effluent, Halogens	N/A	EE, SS
510C	E3	Common Plant Vent	Radiation, Effluent, Flow Rate	SV3	
510D	E3	Admin Bldg Exhaust Vent	Radiation, Effluent, Particulates From The 4th Floor	N/A	DD, OO
510E	E3	Admin Bldg Exhaust Vent	Radiation, Effluent, Halogens From The 4th Floor	N/A	DD, OO
510F	E3	Admin Bldg Exhaust Vent	Radiation, Effluent, Flow Rate, 4th Floor	NONE	DD, OO

REG GUIDE 1.97 STATUS OF COMPLIANCE **NOTES**

NOTE PP: As per Regulatory Guide 1.97 Rev. 3, seismic qualification is not required for Category II variables.

NOTE QQ: Instrumentation identification will be addressed in conjunction with our Control Room Design Review Program in order to provide a coordinate Human Factors Approach.

NOTE RR: No longer required as per Rev. 3 of Regulatory Guide 1.97.

NOTE SS: If the plant vent sampling capability, the wide-range vent monitor , or the main steam line radiation monitor is inoperable in MODES 1, 2, or 3, initiate a pre-planned alternate sampling / monitoring capability as soon as practical, but no later than 72 hours after identification of the failure.