Northeast
Nuclear Energy

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The Northeast Utilities System

JUN 30 2000

Docket Nos. 50-336
50-423

B18158

RE: 10 CFR 50.71(e)

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

Millstone Nuclear Power Station, Unit Nos. 2 and 3
Unit No. 2 Final Safety Analysis Report - Revision 18 and
Unit No. 3 Final Safety Analysis Report - Revision 13

In accordance with 10 CFR 50.71(e), Northeast Nuclear Energy Company (NNECO) hereby submits the following enclosed revisions updating the Unit Nos. 2 and 3 Final Safety Analysis Reports (FSAR):

Enclosure 1	Unit 2 Final Safety Analysis Report - Revision 18
Enclosure 2	Unit 3 Final Safety Analysis Report - Revision 13

This submittal covers, for each unit, the period from January 1, 1999, through December 31, 1999. The previous revisions, Revision 17 to the Unit No. 2 FSAR was submitted on June 30, 1999,[1] and Revision 12 to the Unit No. 3 FSAR was submitted on June 29, 1999.[2]

As stipulated in 10 CFR 50.71(e)(1), each enclosure is provided in the form of replacement-pages together with a List of Effective Pages that identify the current FSAR pages for each unit.

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(1) R. P. Necci letter to the U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 2, Revision 17 to the Final Safety Analysis Report and Addendum 6 to the Annual Report," dated June 30, 1999 (B17819).

(2) R. P. Necci letter to the U.S. Nuclear Regulatory Commission, "Millstone Nuclear Power Station, Unit No. 3, Revision 12 to the Final Safety Analysis Report And Addendum 3 to Annual Report," dated June 29, 1999 (B17767).

A053 1/11

As required by 10 CFR 50.71(e)(2)(ii), "This submittal shall include an identification of changes made under the provisions of Section 50.59 but not previously submitted to the Commission." Accordingly, the safety evaluations for the FSAR changes in this submittal have been included within the 10 CFR 50.59 annual report for each unit, which is being transmitted concurrently with this submittal.

Pursuant to 10 CFR 50.71(e)(2)(i), I certify that this revision accurately presents changes completed since our previous submittal that are necessary to reflect information and analyses submitted to the Commission or prepared pursuant to Commission requirements, as of December 31, 1999.

There are no regulatory commitments contained within this letter.

If you have any questions concerning this submittal, please contact Mr. David A. Smith at (860) 437-5840.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

Raymond P. Necci
Vice President - Nuclear Technical Services

Enclosures (2): Unit No. 2 Final Safety Analysis Report - Revision 18
Unit No. 3 Final Safety Analysis Report - Revision 13

cc: H. J. Miller, Region I Administrator
J. I. Zimmerman, NRC Project Manager, Millstone Unit No. 2
D. P. Beaulieu, Senior Resident Inspector, Millstone Unit No. 2
V. Nerses, NRC Senior Project Manager, Millstone Unit No. 3
A. C. Cerne, Senior Resident Inspector, Millstone Unit No. 3

Docket Nos. 50-336
50-423

B18158

Enclosure 1

Millstone Nuclear Power Station, Unit No. 2

Final Safety Analysis Report - Revision 18

MILLSTONE UNIT NO. 2 - FINAL SAFETY ANALYSIS REPORT
INSERT INSTRUCTIONS, REVISION 18, JUNE 30, 2000

Location (After)	Remove	Insert	Approved in FSARCR Number
*Indicates foldout tables and figures/P&IDs located in back of change package			
		Volume I	
Frontispiece	EP1-1 thru EP14-4	EP1-1 thru EP14-4	Administrative
Table of Contents Tab	DF-1 thru DF-4 FD-1 thru FD-4	DF-1 thru DF-4 FD-1 thru FD-4	Administrative Administrative
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Chapter 3 Tab	3-i thru 3-vii	3-i thru 3-vii	Administrative
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INSERT INSTRUCTIONS, REVISION 18, JUNE 30, 2000

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Section 7.5 Tab	7.5-26	7.5-26	99-86
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13.9-1 through 13.9-2	07/93
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14.5-1	10/94
14.6-1 through 14.6-15	02/99
14.6-16	04/99
14.6-17 through 14.6-22	02/99
14.6-23 through 14.6-29	03/99
14.7-1	04/99
14.7-2 through 14.7-5	04/93
14.8-1	08/99
14.8-2 through 14.8-5	04/99
14.8-6	09/99
14.8-7	04/99
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14.8-12 through 14.8-14	04/99
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T14.2.2-1 through T14.2.2-2	09/98
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T14.4.3-1 (Deleted)	10/98
T14.4.3.1-1 through T14.4.3.1-3	10/98
T14.4.3.5-1 through T14.4.3.5-2	10/98
T14.4.4-1 through T14.4.4-2	10/98
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T14.4.6-2	07/98
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F14.0.7-8 (Deleted)	04/93
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F14.1.5-1 through F14.1.5-13 (Deleted)	03/99
F14.1.5.1-1 through F14.1.5.1-11	03/99
F14.1.5.2-1 through F14.1.5.2-16	03/99
F14.2.1-1 through F14.2.1-17	09/98
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F14.2.7-1 through F14.2.7-10	04/99
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F14.4.8-10 through F14.4.8-12	04/93
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10003		01.02-02	01
13013	10	09.08-06	01
13013	14	09.08-03	01
13013	19	09.08-04	01
17001		11.02-01	01
17002		11.02-02	01
17003		11.02-03	01
17004		11.02-04	01
17005		11.02-05	01
17006		11.02-06	01
17007		11.02-07	01
17008		11.02-08	01
17009		11.02-09	01
17010		11.02-10	01
20104		06.04-01	01
26001	01	09.00-01	01
26001	02	09.00-02	01
26002	01	10.03-01	01
26002	02	10.03-01	02
26002	03	10.03-01	03
26002	04	10.03-01	04
26005	01	10.04-02	01
26005	02	10.04-02	02
26005	03	10.04-02	03
26005	04	10.04-02	04
26007	01	09.07-02	01
26007	02	09.07-02	02
26007	03	09.07-02	03
26008	01	09.07-01	01
26008	02	09.07-01	02
26008	03	09.07-01	03
26008	04	09.07-01	04
26009	01	09.11-01	01
26010	01	08.03-04	01
26011	01	09.10-02	01
26011	02	09.10-02	02
26012		10.04-01	01
26014	01	04.01-01	01
26014	02	04.01-01	02
26014	03	04.01-01	03
26014	03	04.03-04	01
26015	01	06.01-01	01
26015	02	06.01-01	02
26015	03	06.01-01	03
26017	01	09.02-02	01
26017	02	09.02-02	02
26017	03	09.02-02	03
26018	02	08.03-05	02
26018	03	08.03-05	03
26018	04	08.03-05	01A
26018	05	08.03-05	01B
26020	01	11.01-01	01
26020	02	11.01-02	01
26021	01	11.01-04	01
26021	02	11.01-05	01
26022	01	09.04-01	01

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NUSCO DRAWING NUMBER	DWG SHEET NUMBER	FSAR FIGURE NUMBER	FIGURE SHEET NUMBER
26022	02	09.04-02	01
26022	03	09.04-03	01
26022	04	09.04-04	01
26022	05	09.04-05	01
26022	06	09.04-06	01
26023	01	09.05-01	01
26023	02	09.05-01	02
26024	01	11.01-03	01
26024	02	11.01-03	02
26024	03	11.01-03	03
26025	01	09.06-01	01
26025	02	09.06-02	01
26025	03	09.06-03	01
26026	01	09.13-01	01
26026	02	07.10-01	01
26026	02	09.13-01	02
26026	03	08.03-04	02
26026	03	09.13-01	03
26026	04	08.03-04	03
26026	04	09.13-01	04
26027	01	09.09-04	01
26027	02	09.09-31	01
26027	03	09.09-04	0A
26028	01	09.09-01	01
26028	02	09.09-02	01
26028	03	09.09-02	02
26028	04	09.09-02	03
26028	05	09.09-02	04
26029	01	09.09-03	01
26029	02	09.09-03	02
26029	03	09.09-03	03
26030	01	09.12-01	01
26074	01	09-06-04	01
26074	02	09-06-05	01
27011		01.02-03	01
27012		01.02-04	01
27013		01.02-05	01
27014		01.02-06	01
27015		01.02-07	01
27016		01.02-08	01
27017		01.02-09	01
27018		01.02-10	01
27019		01.02-11	01
27020		01.02-12	01
27021		01.02-13	01
27022		01.02-14	01
27023		01.02-15	01
27024		01.02-16	01
27025		01.02-17	01
27027		07.06-01	01
28006	01	07.06-09	01
28006	02	07.06-10	01
28129	22	09.09-10	01
28105	36	07.09-01	01
28150	01	07.03-01	01
28150	02A	07.03-02A	01

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NUSCO DRAWING NUMBER	DWG SHEET NUMBER	FSAR FIGURE NUMBER	FIGURE SHEET NUMBER
28150	02B	07.03-02B	01
28150	02C	07.03-02C	01
28150	02D	07.03-02D	01
28150	03	07.03-03	01
28150	04	07.03-04	01
28202	02	07.04-05	01
29155	09	07.04-03	01
29192	01	07.04-01	01
29193	01	07.02-02	01
29198	14	07.02-08	01
30001		08.02-01	01
30023		08.05-01	02
30024		08.05-01	01
30034	01	08.03-01	0A
30034	02	08.03-01	0B
30071		07.06-11	01
30208	01	07.06-02	01
30208	02	07.06-02	02
30208	03	07.06-02	03
30208	04	07.06-02	04
30209	01	07.06-03	01
30209	02	07.06-03	02
30210	01	07.06-03	03
30210	02	07.06-03	04
30211	01	07.06-04	01
30211	02	07.06-04	02
30211	03	07.06-04	03
30211	04	07.06-04	04
30212	01	07.06-05	01
30212	02	07.06-05	02
30212	03	07.06-05	03
30212	04	07.06-05	04
30213	01	07.06-06	01
30213	02	07.06-06	02
30214	01	07.06-07	01
30214	02	07.06-07	02
30214	03	07.06-07	03
30214	04	07.06-07	04
30215	01	07.06-08	01
30215	02	07.06-08	02
30215	03	07.06-08	03
30215	04	07.06-08	04
32009	42	08.04-01	01
32012	48	07.09-03	01
32013	39	08.04-02	01
32041	01	08.03-02	01
32041	01A	08.03-02	01A
32041	02	08.03-02	02
32041	02A	08.03-02	02A
32041	03	08.03-02	03
32041	04	08.03-02	04
32041	05	08.03-02	05
32041	06	08.03-02	06
32041	07	08.03-02	07
32041	08	08.03-02	08
32041	09	08.03-02	09

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NUSCO DRAWING NUMBER TO FSAR FIGURE NUMBER MATRIX

NUSCO DRAWING NUMBER	DWG SHEET NUMBER	FSAR FIGURE NUMBER	FIGURE SHEET NUMBER
32041	0A	08.03-02	00A
32041	10	08.03-02	10
32041	11	08.03-02	11
32041	12	08.03-02	12
32041	13	08.03-02	13
32041	14	08.03-02	14
32041	15	08.03-02	15
32041	16	08.03-02	16
32041	17	08.03-02	17
32041	18	08.03-02	18
32041	19	08.03-02	19
32041	20	08.03-02	20
32041	21	08.03-02	21
32041	22	08.03-02	22
32041	23	08.03-02	23
32041	24	08.03-02	24
32041	25	08.03-02	25
32041	26	08.03-02	26
32041	27	08.03-02	27
32041	28	08.03-02	28
32041	29	08.03-02	29
32041	30	08.03-02	30
32045	01	08.05-02	01
32045	02	08.05-02	02
32045	03	08.05-02	03
32045	04	08.05-02	04
32045	05	08.05-02	05
32045	06	08.05-02	06
32045	07	08.05-02	07
32045	08	08.05-02	08
32045	09	08.05-02	09
32045	10A	08.05-02	10A
32045	10B	08.05-02	10B
32045	10C	08.05-02	10C
32045	10D	08.05-02	10D
32045	11	08.05-02	11
32045	12	08.05-02	12
34031		08.07-01	02
34032		08.07-01	03
34042		08.07-01	04
34060		08.07-01	01
37005	105	07.09-02	01
39220	07A	07.04-06	01
39220	07B	07.04-06	02
39220	08A	07.04-06	03
39220	08B	07.04-06	04
39220	09A	07.04-06	05
39220	09B	07.04-06	06
39220	10A	07.04-06	07
39220	10B	07.04-06	08
39220	11A	07.04-06	09
39220	11B	07.04-06	10

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FSAR FIGURE NUMBER	FIGURE SHEET NUMBER	NUSCO DRAWING NUMBER	DWG SHEET NUMBER
01.02-02	01	10003	
01.02-03	01	27011	
01.02-04	01	27012	
01.02-05	01	27013	
01.02-06	01	27014	
01.02-07	01	27015	
01.02-08	01	27016	
01.02-09	01	27017	
01.02-10	01	27018	
01.02-11	01	27019	
01.02-12	01	27020	
01.02-13	01	27021	
01.02-14	01	27022	
01.02-15	01	27023	
01.02-16	01	27024	
01.02-17	01	27025	
04.01-01	01	26014	01
04.01-01	02	26014	02
04.01-01	03	26014	03
04.03-04	01	26014	03
06.01-01	01	26015	01
06.01-01	02	26015	02
06.01-01	03	26015	03
06.04-01	01	20104	
07.02-02	01	29193	01
07.02-08	01	29198	14
07.03-01	01	28150	01
07.03-02A	01	28150	02A
07.03-02B	01	28150	02B
07.03-02C	01	28150	02C
07.03-02D	01	28150	02D
07.03-03	01	28150	03
07.03-04	01	28150	04
07.04-01	01	29192	01
07.04-03	01	29155	09
07.04-05	01	28202	02
07.04-06	01	39220	07A
07.04-06	02	39220	07B
07.04-06	03	39220	08A
07.04-06	04	39220	08B
07.04-06	05	39220	09A
07.04-06	06	39220	09B
07.04-06	07	39220	10A
07.04-06	08	39220	10B
07.04-06	09	39220	11A
07.04-06	10	39220	11B
07.06-01	01	27027	
07.06-02	01	30208	01
07.06-02	02	30208	02
07.06-02	03	30208	03
07.06-02	04	30208	04
07.06-03	01	30209	01
07.06-03	02	30209	02
07.06-03	03	30210	01
07.06-03	04	30210	02
07.06-04	01	30211	01

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FSAR FIGURE NUMBER	FIGURE SHEET NUMBER	NUSCO DRAWING NUMBER	DWG SHEET NUMBER
07.06-04	02	30211	02
07.06-04	03	30211	03
07.06-04	04	30211	04
07.06-05	01	30212	01
07.06-05	02	30212	02
07.06-05	03	30212	03
07.06-05	04	30212	04
07.06-06	01	30213	01
07.06-06	02	30213	02
07.06-07	01	30214	01
07.06-07	02	30214	02
07.06-07	03	30214	03
07.06-07	04	30214	04
07.06-08	01	30215	01
07.06-08	02	30215	02
07.06-08	03	30215	03
07.06-08	04	30215	04
07.06-09	01	28006	01
07.06-10	01	28006	02
07.06-11	01	30071	
07.09-01	01	28105	36
07-09-02	01	37005	105
07.09-03	01	32012	48
07.10-01	01	26026	02
08.02-01	01	30001	
08.03-01	0A	30034	01
08.03-01	0B	30034	02
08.03-02	00A	32041	0A
08.03-02	01	32041	01
08.03-02	01A	32041	01A
08.03-02	02	32041	02
08.03-02	02A	32041	02A
08.03-02	03	32041	03
08.03-02	04	32041	04
08.03-02	05	32041	05
08.03-02	06	32041	06
08.03-02	07	32041	07
08.03-02	08	32041	08
08.03-02	09	32041	09
08.03-02	10	32041	10
08.03-02	11	32041	11
08.03-02	12	32041	12
08.03-02	13	32041	13
08.03-02	14	32041	14
08.03-02	15	32041	15
08.03-02	16	32041	16
08.03-02	17	32041	17
08.03-02	18	32041	18
08.03-02	19	32041	19
08.03-02	20	32041	20
08.03-02	21	32041	21
08.03-02	22	32041	22
08.03-02	23	32041	23
08.03-02	24	32041	24
08.03-02	25	32041	25
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08.05-02	03	32045	03
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The protective system is isolated from the control instrumentation systems so that failure or removal from service of any control instrumentation system component or channel does not inhibit the function of the protective system.

1.4.7 Electrical Systems

Normal, reserve and emergency sources of auxiliary electrical power are provided to assure safe and orderly shutdown of the plant and to maintain a safe shutdown condition under all credible circumstances. Onsite electrical power sources and systems are designed to provide dependability, independence, redundancy and testability in accordance with the requirements of 10 CFR Part 50, Appendix A. The load-carrying capability and other electrical and mechanical characteristics of emergency power systems are in accordance with the requirements of Safety Guide No. 9. Two redundant, independent, full-capacity emergency power sources and distribution subsystems are provided. Each of these subsystems powers all equipment in the associated safety related subsystems as described in Subsection 1.4.5.

1.4.8 Radioactive Waste Processing System

The radioactive waste processing system (see Section 11.1) is designed so that discharges of radioactivity to the environment are minimized and are in accordance with the requirements of Sections 105 and 106 and Appendix B of 10 CFR Part 20 (version prior to January 1, 1994) and Appendix I of 10 CFR Part 50.

1.4.9 Radiation Protection

Millstone Unit 2 is provided with a centralized control room which has adequate shielding (see Section 11.2.2.3) and ventilation system features (see Section 9.9.10) to permit occupancy during all postulated accidents involving radiation releases. 99-91

The radiation shielding in Millstone Unit 2 and the radiation control procedures ensure that operating personnel do not receive exposures during normal operation and maintenance in excess of the applicable limits of 10 CFR Part 20.

1.4.10 Fuel Handling and Storage

Fuel handling and storage facilities (see Section 9.8) are provided for the safe handling and storage of fuel. The design precludes accidental criticality.

A. Fuel-bundle inlet flow blockage during full-power operation and subsequent overheating of the coolant-starved fuel, or

B. loss of reactor coolant.

Condition A, inlet flow blockage during full-power operation and subsequent overheating and melting of the fuel, is not considered possible because open (nonshrouded) fuel bundles are used, thereby providing cross-flow to the flow-starved channel even if some of the inlet holes were blocked. Details and conclusions of the tests performed at Combustion Engineering (CE), Inc. on the influence of inlet geometry on flow in the entrance region are presented in ASME paper 68-WA/HT-34 delivered at the December 1968 Winter Annual Meeting. Further analysis of these tests showed that if a group of four flow holes in the core support plate at the base of the fuel bundle were blocked, the subchannels above the blocked region would have an inlet velocity about 21 percent of the core average bulk inlet velocity. Because of crossflow from the surrounding nonblocked regions, the net effect of this flow shortage, using conservative calculations, is to increase the enthalpy rise of the blocked region by a maximum of 35 percent. At nominal conditions, the hot channel departure from nucleate boiling ratio (DNBR) would drop from 2.0 to 1.4, assuming that the blockage occurred directly below the design hot channel.

Condition B was covered comprehensively in the Statement of Affirmative Testimony and Evidence of Combustion Engineering in the Matter of Rulemaking Hearing for the Acceptance Criteria for Emergency Core Cooling System for Light-Water-Cooled Nuclear Power Reactors, Docket No. RM-50-1. The emergency core cooling system (ECCS) is designed to remove the decay heat from the core for the necessary period of time following a loss-of-coolant accident (LOCA). Core power distributions and LOCA temperature-time histories indicate that for peak clad temperatures below 2300°F, the total clad oxidation will be significantly less than 1 percent.

1.8.1.3 Primary System Quality Assurance and In-Service Inspectability

The Northeast Utilities Service Company (NUSCO) has traditionally retained full responsibility and maintained close control over all aspects of the design and construction of its power plants. This background of experience was used by NUSCO in establishing a comprehensive quality assurance program to assure that the Millstone Unit 2 is designed, fabricated, and constructed in accordance with the requirements of applicable specifications and codes. The NUSCO program started with the initial plant design and has continued through all phases of equipment procurement, fabrication, erection, construction, and plant operation. The program provides for review of specifications to assure that quality control requirements are included and for surveillance and audits of the manufacturing and construction efforts to assure that the specified requirements are met.

A summary description of the NUSCO Quality Assurance Program (NUQAP) is included as Section 12.8. This program fully meets the guidelines established in the former AEC Regulation 10 CFR Part 50, Appendix B entitled "Quality Assurance Criteria for Nuclear Power Plants." The quality assurance organization is described in The NUQAP Topical Report. That information is incorporated herein by reference. 99-78

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

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

All structures are designed in accordance with accepted and time proven building codes (as specified in Subsection 5.1.2) for the loading conditions stated in Subsections 5.2.3, 5.3.3, 5.4.3, 5.5.3 and 5.6.3 which insures that they will operate under normal conditions in a safe manner. In addition, those structures and/or components which could affect public safety were designed to function safely during an earthquake as discussed in Section 5.8. Wind and tornado storm protection design criteria are discussed in Subsections 5.2.3.1.6, 5.3.3.1.4, 5.4.3.1.6, 5.5.3.3.2, 5.6.3.1.5, and 5.7.3.1.4. Protection against postulated missiles is discussed in Section 5.2.6.1.

The design loads for the containment and major component supports to ensure a safe shutdown after a loss-of-coolant accident are described in Subsection 5.2.3.1.3.

Systems and components important to safety are designed to operate satisfactorily and to be compatible with environmental conditions associated with normal operation and postulated accident conditions. Those systems and components located in the containment are designed to operate in an environment of 289°F and 54 psig. Systems and components important to safety are designated Seismic Class I and designed in accordance with the criteria given in Subsection 5.2.5.3. Missile protection and pipe whipping protection criteria for these systems and components are given in Subsections 5.2.6.1 and 5.4.3.1.

Leak-before-break (LBB) analyses for the reactor coolant system (RCS) main coolant loops, for the pressurizer surge line, and unisolable RCS portions of the safety injection and shutdown cooling piping, which demonstrated that the probability of fluid system piping rupture was extremely low, was reviewed and approved by the commission. Accordingly, pursuant to GDC 4, 1998 revision, the dynamic effects associated with pipe ruptures in the above piping segments, including the effects of pipe whipping and discharging fluids have been excluded from the design basis of the following components and systems: 99-71

Core barrel snubbers, core barrel stabilizer blocks
Reactor vessel core support ledge
Neutron shield tank,
Protection of Closed Systems
- RBCCW piping
- SGBD piping
- SGBD sampling piping

CRITERION 50 - CONTAINMENT DESIGN BASIS

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

The containment structure, including the access openings, penetrations and the containment heat removal system, is designed to withstand a pressure of 54 psig and a temperature of 289°F following a loss-of-coolant accident (LOCA) or a main steam line break accident (see Section 14.8.2). Details of the methods used to analyze the containment structure are described in Subsection 5.2.3. To obtain an adequate margin of safety, a factored load was selected for a design which allows a 25 percent increase over the calculated postulated accident load. 99-73

A high degree of leak tightness is provided by a one-quarter inch thick steel liner plate which completely encloses the interior surface of the containment structure. Components of the liner plate, such as penetration sleeves, personnel locks, and equipment hatch, are designed to meet the requirements of the 1968 ASME Boiler and Pressure Vessel Code, Section III (Nuclear Vessels) Paragraph N-1211. Further description of the liner plate is contained in Subsection 5.2.4.

As a further check on the design a structural integrity test, composing a test pressure load of 115 percent of the design accident pressure load, is conducted prior to operation. In addition to this, a leak rate test will be conducted prior to operation and at certain intervals during operation. Details of the leak rate test are provided in Subsection 5.2.9.1.

2. Travelling screens are placed on continuous wash at high speed.

3. Tours are made within the fenced boundary of the site to ensure all loose material is removed or thoroughly tied down.

4. Higher plant supervision is notified of the storm warning.

5. Should a sustained wind speed of 50 MPH be measured at the meteorological tower, the following will be accomplished:

   a. Both emergency power sources will be started and left idling.

   b. Additional personnel will be called in to assist plant operators, as required.

6. Should sustained water level approach grade level due to extremely high tides, not wave action, plant management will assess the general plant condition in conjunction with the Millstone Unit 2 PORC. Typical items to be discussed would be the intensity of the storm as to whether it is building, static, or waning; status of all plant equipment and buildings; and status of any building leakage. At this time, the decision will be made as to whether to continue operations or to initiate a normal plant shutdown. In either case, Directorate of Regulatory Operations, Region 1, will be notified of plant condition and management decision.

7. One service water pump motor will be secured and protected against flooding to a minimum elevation of 28 ft. MSL in accordance with technical specifications.

8. If the severity of the hurricane is such that the sustained water level is increasing above grade level:

   a. Disable diesel fuel oil transfer pumps to prevent transfer of water from underground storage tank to supply tanks.

   b. Station an operator inside the intake structure to monitor the intake structure water level with respect to the Service Water and Circulating Water Pump Motors. 99-79

9. As long as the 345 transmission lines remain uninterrupted and plant equipment is not endangered, the plant can continue to be operated at power until the level of water in the intake structure reaches 19.5 ft. MSL which is just below the level of the Circulating Water Pump Motors. At this time, the plant would be tripped and placed in a hot standby condition in accordance with plant procedures. If outside power is lost, emergency loads would be supplied by the diesels, and additional operator response would be in accordance with plant procedures.

10. If, following the securing of the circulating water pumps, the level of water within the Intake Structure rises above 19.5 ft. MSL, one diesel would be immediately secured and its cooling system realigned such that the diesel can be cooled by water supplied from the fire system in accordance with applicable operating procedures.

11. If the water level within the Intake Structure rises to 22 ft. MSL, thereby endangering the service water pump motors, the diesel operating on service water would be secured and the Service Water Pumps would be tripped in accordance with plant procedures. Essential plant equipment would be powered by the diesel whose cooling system had previously been modified to use fire water. The plant would be maintained in a hot standby status. Decay heat would be transferred from the core to the steam generators by natural circulation and then be removed from the steam generators by steaming through the atmospheric dumps. The generators would be fed by either a steam-driven or electric auxiliary feedwater pump taking a suction on the condensate storage tank.

12. When the water level recedes below 22 ft. MSL, the motor which was protected would be recommissioned and started in accordance with plant procedures. It would then be used to cool the diesel which had not been lined up to the fire system. The time required for this step once the water level has receded is approximately 2.0 hours.

13. The other service water pump motor(s) would then be removed, disassembled, steam cleaned, dried, reassembled and reinstalled, and then an additional pump would be started. This will then allow plant systems and equipment to be returned to a normal hot standby status.

14. When the water level recedes below grade elevation 14'-6", sample/remove/verify no water in underground diesel fuel oil storage tank, then reenable diesel fuel oil transfer pumps. 99-79

B. Justification

By the actions described, plant personnel can readily maintain the plant in a safe shutdown condition through the PMH where the major concern is the removal of decay heat. The feasibility and reliability of this course of action is justified as follows:

1. Decay heat is transferred from the core to the steam generators by natural circulation and then is removed from the steam generators by steaming through the atmospheric dumps. The generators are fed by either a steam-driven or an electric auxiliary feedwater pump taking a suction on the condensate storage tank. The availability of this method of decay heat removal is justified as follows:

a. Although as previously described, plant personnel have taken action to ensure the continued operation of at least one diesel (and this is further justified in subsequent paragraphs), the incorporation of a steam-driven pump and manually-positionable components into the plant design provides for decay heat removal by the method described without dependence on emergency power from the diesels.

b. Although the normal volume in the Condensate Storage Tank is at least 225,000 gallons, the Technical Specification minimum level of 165,000 gallons is sufficient to remove decay heat for a minimum period of 10 hours. As a reserve supply for the auxiliary feedwater system, the suction of the pumps can be supplied from the two 245,000 gallon fire water storage tanks (Section 9.10.2). Consequently, the 10 hours mentioned can be readily extended. Further, these tanks are supplied by the city water system which is expected to remain pressurized by diesel-driven pumps, thereby providing a virtually unlimited supply of water. Finally, the extended periods of decay heat removal provided by the sources already mentioned would provided time to restore power to the primary water transfer pumps, thereby making available the additional 100,000 gallons from the Primary Water Storage Tank although it is not felt that this would be required. (98-5)

c. The two electric and one steam-driven auxiliary feedwater pumps are operationally checked quarterly to ensure their availability.

2. Although not required for decay heat removal, the emergency power from one diesel generator is maintained by shifting its cooling to the fire system. The availability of this alternate cooling is ensured as follows:

a. The fire system is pressurized by one electric-driven fire pump from Unit 1, one electric-driven fire pump from Unit 2 and a diesel-driven fire pump. The Unit 1 electric pump can be aligned from the Unit 1 Gas Turbine. These pumps are performance tested monthly to ensure their operability.

b. For the postulated storm, the fire pump house is located in the lee of the Millstone Units 1 and 2 building complex. The flood protection of this pump house to a level of 22.33 ft. MSL is, therefore, sufficient to protect the pumps against the maximum PMH standing water level of 18.11 ft MSL. The Millstone Unit 1 Gas Turbine is also protected to a level of 19 ft. MSL.

c. The fire water storage tanks are maintained full. The water supply is estimated to be sufficient to operate one diesel, supplying hot standby loads, for at least 8 hours. (98-127)

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3A-13	SAMMSOR-DYNASOR Finite Element Model of Core Support Barrel

3.4.3.1 Analytical Methodology

The methods used in the Cycle 13 core analysis are described in References 3.4-4 through 3.4-8. In summary, the reference neutronic design analysis of the reload core was performed using the XTGPWR (Reference 3.4-9) reactor simulator code. The input isotopics data was based on quarter core depletion calculations performed for Cycle 13 using the XTGPWR code. The CASMO/MICBURN assembly depletion model (Reference 3.4-8) generated the cross section input to the XTGPWR model. Local peaking factors used in XTGPWR are based on quarter core PDQ/HARMONY calculations for the Cycle 13 loading plan. The fuel shuffling between cycles was accounted for in the calculations.

Calculated values of LHR and F_r were determined with the XTGPWR reactor model. The calculational thermal-hydraulic feedback and axial exposure distribution affects on power shapes, rod worths, and cycle lifetime are explicitly included in the analysis.

3.4.3.2 Physics Characteristics

The neutronics characteristics of the Cycle 13 core are presented in Table 3.4-2. The safety analysis for Cycle 13 is applicable for Cycle 12 burnups of 15,115 to 16,615 MWD/MTU and a Cycle 13 burnup of up to 19,467 MWD/MTU. The boron letdown curve for Cycle 13 is shown in Figure 3.4-4. The BOC 13 HZP, ARO xenon free critical boron concentration is calculated to be 1453 ppm. At 150 MWD/MTU, equilibrium xenon, hot full power (HFP) ARO, the critical boron concentration is calculated to be 1024 ppm. The soluble boron is projected to reach 0 ppm at a Cycle 13 length of 17,698 MWD/MTU at a core power of 2,700 MWt. 99-70

99-70

3.4.3.2.1 Power Distribution Considerations

Representative calculated power maps for Cycle 13 are shown in Figures 3.4-5 and 3.4-6 for BOC (equilibrium xenon), and EOC conditions, respectively. The power distributions were obtained from a three-dimensional XTGPWR model with moderator density and Doppler feedback effects incorporated.

As shown in Figure 3.4-5 for the design Cycle 13 loading pattern, the calculated BOC HFP equilibrium xenon nuclear peaking factor F_r is 1.600. The peak LHR, including uncertainties at BOC, HFP is calculated to be 12.8 kW/ft. At EOC conditions, the corresponding values for F_r and LHR are 1.517 and 11.6 kW/ft, respectively. The technical specification limits on F_r and LHR are 1.69 and 14.6 kW/ft, respectively.

3.4.3.2.2 Control Rod Reactivity Requirements

Shutdown margin evaluations for Cycle 13 are given in Table 3.4-3. The Millstone Unit 2 technical specifications require a minimum shutdown margin of 3,600 pcm. The Cycle 13 excess shutdown margin from HFP is 124 pcm at BOC and 323 pcm at EOC. The excess shutdown margin from hot zero power (HZP) is 140 pcm at BOC and 751 pcm at EOC.

3.4.3.2.3 Moderator Temperature Coefficient Considerations

The technical specifications require that the MTC be less than +7 pcm/°F at or below 70 percent of rated thermal power, less than +4 pcm/°F above 70 percent power and greater than -28 pcm/°F at 100 percent of rated thermal power. The BOC HZP all rods out (ARO) MTC for the Cycle 13 core is calculated to be +2.0 pcm/°F assuming an EOC 12 burnup of 15,615 MWD/MTU, which is within the technical specification limit at HZP conditions. The MTC at 100 percent rated power, BOC, equilibrium xenon conditions is calculated to be -6.0 pcm/°F. The MTC at EOC, HFP, equilibrium xenon conditions is calculated to be -23.3 pcm/°F.

3.4.4 Post-Reload Startup Testing

Startup tests will be performed at the beginning of each reload cycle to obtain the as-built core characteristics and to verify technical specification and core physics design parameters.

As a minimum, the reload startup physics test program shall consist of the following:

a. Critical Boron Concentration - HZP, Control Rods Withdrawn.

b. Critical Boron Concentration - HZP, Control Rod Group(s) of at least 1% reactivity are fully inserted in the core.

c. Control Rod Group Worths - HZP, two or more control rod groups shall be measured which are well distributed radially and represent a predicted total worth of at least 3% reactivity.

d. Isothermal Temperature Coefficient - HZP.

e. Flux Symmetry - between 0 and 30% of full power.

f. Power Distribution - between 40 and 75% of full power.

g. Isothermal Temperature Coefficient - greater than 70% of full power.

h. Power Distribution - greater than 90% of full power.

i. Critical Boron Concentration - greater than 90% of full power.

The above "reload startup physics test program (steps a. through i.)" was applicable for the startup testing of a normal beginning of cycle startup after a refueling outage. The following test program for the restart of Cycle 13 after the extended shutdown will be performed. The reactor core has been reloaded successfully and verified per procedure after a full core offload. Performance of any low power physics testing is not required. However, the following tests are performed in order to ensure that fuel misloading has not occurred, Control Element Assemblies (CEA) are latched to their extension shafts, and that Technical Specification requirements are satisfied: 99-70

1) CEA Extension Shaft Weight Measurement,
2) Individual CEA Drop Time Measurement,
3) Flux Symmetry Measurement at 30% power,
4) Incore Detector Operability Surveillance at 30% power, and 99-70
5) Radial Peaking Factor Surveillance at 30% power.

3.4.5 Reactor Stability

3.4.5.1 General

Xenon induced spatial oscillations on the Millstone Unit 2 core fall into three classes or modes. These are referred to as axial oscillations, azimuthal oscillations, and radial oscillations. An axial oscillation is one in which the axial power distribution periodically shifts to the top and bottom of the core. An azimuthal oscillation is one in which the X-Y power distribution periodically shifts from one side of the core to the other. A radial oscillation is one in which the X-Y power distribution periodically shifts inward and outward from the center of the core to the periphery.

Xenon stability analyses indicate that a number of general statements can be made:

a) The time scale on which the oscillations occur is long, and any induced oscillations typically exhibit a period of 25 to 30 hours. (98-1)

b) As long as the initial power peaking associated with the perturbation initiating the oscillation is within the limiting conditions for operation, specified acceptable fuel design limits will not be approached for a period of hours allowing an operator time to decide upon and take appropriate remedial action prior to the time when allowable peaking factors would be exceeded.

c) The core will be stable to radial mode oscillations at all times in the burnup cycle.

d) The core will be stable to azimuthal mode oscillations at all times in the burnup cycle.

e) All possible modes of undamped oscillations can be detected by both excore and in-core instrumentation as discussed below.

3.4.5.2 Detection of Oscillations

Primary reliance for the detection of any xenon oscillations is placed on the excore flux monitoring instrumentation. The power range excore neutron detectors (one axial pair per quadrant) are used to monitor the symmetry of power distributions and are located at distinct azimuthal and axial positions. These detectors are sensitive primarily to the power density variations produced by peripheral fuel assemblies in the vicinity of the detectors. All possible xenon induced spatial oscillations will affect the power densities of the peripheral fuel assemblies in the core.

In addition, the in-core instrumentation provides information which will be used in the early stages of cycle operation to confirm predicted correlations between indications from the excore detectors and the space-dependent flux distribution within the core. Later on, during normal operation, the in-core detector system provides information which may be used to supplement that available from the excore detectors.

3.4.5.3 Control of Oscillations

Since the reactor will not be operated under conditions that imply instability with respect to azimuthal xenon oscillation, no special protective system features are needed to accommodate asimuthal mode oscillations. Regardless, a maximum azimuthal power tilt is prescribed in the Technical Specifications along with prescribed operating restrictions in the event that the azimuthal power tilt limit is exceeded.

As described earlier, the power range excore neutron detectors are used to monitor the azimuthal symmetry of the power distributions since they are located at distinct locations in the X-Y plane. Should the excore detectors indicate different readings in the azimuthal direction, a tilt in the core power distribution would be indicated. When the tilt exceeds a preset magnitude an alarm will occur. In the event of an alarm, the orientation of the tilt will be determined and, on the basis of orientation, the proper CEA's will be manually adjusted to reduce the magnitude of the tilt.

The features provided for azimuthal xenon oscillation control are:

a) instrumentation for monitoring azimuthal power tilt. (98-1)

b) administrative limits on azimuthal power tilt.

The excore detectors are used to monitor the axial power distribution and to detect deviations from the equilibrium distribution such as those which would occur during an axial xenon oscillation. This is done by monitoring variations in the external axial shape index, a parameter derived from the excore detector readings which is related to the axial power distribution. Control of axial xenon oscillation is accomplished utilizing Regulating Bank 7. When it is determined that the axial shape index may exceed the boundaries of a specified control band about the equilibrium value, this bank is slowly inserted and eventually withdrawn over a period of several hours. The core is then stabilized until a new oscillation develops.

The features provided for axial xenon control and protection are:

a) equipment for monitoring axial shape index.

b) administrative limits on axial power distribution, external axial shape index.

c) an axial shape index reactor trip (local power density - high).

d) use of Regulating Bank 7 for control of axial power distribution.

3.4.5.4 Operating Experience

Recent core designs for Millstone Unit 2 (Cycles 10 and beyond) have been developed to include longer fuel cycles along with low radial leakage fuel management. These current designs scatter load fresh fuel assemblies throughout the interior of the core with the highest burnup fuel assemblies being loaded along the core periphery. Core designs prior to Cycle 10 operation were not of a low radial leakage design due to the loading of fresh fuel assemblies along the core periphery.

With respect to xenon oscillations in the radial and azimuthal directions, studies indicate that core designs of a low radial leakage design (i.e., highest burnup assemblies loaded on the core periphery with fresh fuel assemblies scatter loaded about the core interior) are more stable than those designs which load fresh fuel assemblies along the core periphery. Therefore, the conclusions regarding xenon oscillations in the radial and azimuthal directions, which are presented in Section 3.4.5.5, remain applicable to current plant operations.

With regard to axial xenon oscillations, the core near end-of-cycle may be naturally unstable in the absence of any control rod action even if low leakage core designs are utilized. But axial xenon oscillations are sufficiently slow (the period of oscillation being 25-30 hours) so that there would be sufficient time to control the oscillations. In addition, automatic protection is provided if operator action is not taken to remedy the situation. Regulating Bank 7 CEA's are utilized for controlling axial xenon oscillations.

3.4.5.5 Method of Analysis (98-1)

The classic method for assessing spatial xenon oscillations is that developed by Randall and St. John (Reference 3.4-10) which consists of expanding small perturbations of the flux and xenon concentrations about equilibrium values in eigenfunctions of the system with equilibrium xenon present. However, it is necessary to extend this simple linear analysis to treat cores which are nonuniform because of fuel zoning, depletion, and CEA patterns, for example. Such extensions have been worked out and are reported in References 3.4-11 and 3.4-13. In this extension, the eigenvalue separations between the excited state of interest and the fundamental are computed numerically for symmetrical flux shapes. For nonsymmetrical flux shapes, the eigenvalue separation can usually be obtained indirectly from the dominance ratio λ_1/λ_0, computed during the iteration cycle of the spatial calculation.

Numerical space time calculations are performed in the required number of spatial dimensions for the various modes as checkpoints for the predictions for the extended Randall-St. John treatment described above.

3.4.5.5.1 Radial Xenon Oscillations

To confirm that the radial oscillation mode is extremely stable, a space-time calculation was run for a reflected, zoned core 11 feet in diameter without including the damping effects of the negative power coefficient. The initial perturbation was a poison worth of 0.4 percent in reactivity placed in the central 20 percent in the core for 1 hour. Following removal of the perturbation, the resulting oscillation was followed in 4-hour time steps for a period of 80 hours. Results show that the oscillation died out very rapidly with a

damping factor of about -0.06 per hour. When this damping coefficient is corrected for a finite time mesh by the formula in Reference 3.4-12, it is more strongly convergent. On this basis, it is concluded that radial oscillation instability will not occur.

This conclusion is of particular significance because it means that there is no type of oscillation where the inner portions of the core act independently of the peripheral portions of the core whose behavior is most closely followed by the excore flux detectors. Radial mode oscillations, even though highly damped, would be manifested as periodic variation in the excore flux power signal while the delta-T power signals remained constant. Primary reliance is placed on the excore flux detectors for the detection of any xenon oscillations.

3.4.5.5.2 Azimuthal Xenon Oscillations

Analyses indicate that the eigenvalue separation between the first asimuthal harmonic and the fundamental is about 0.86 percent in λ. The calculated damping coefficient for the first azimuthal mode is -0.016 per hour, and the higher modes will be even more strongly damped. Furthermore, the Doppler coefficient applicable to the Millstone Unit 2 reactor is calculated to be approximately -1.36×10^{-3} $\delta\rho$/(kW/ft) which is sufficiently negative to ensure stability of all the azimuthal modes.

3.4.5.5.3 Axial Xenon Oscillations

As checkpoints for the predictions for the modified Randall-St. John approach, numerical spatial time calculations have been performed for the axial case at both beginning and end-of-cycle. The fuel and poison burnup distributions were obtained by depletion with soluble boron control so that the power distribution was strongly flattened. Spatial Doppler feedback was included in these calculations. The initial perturbation used to excite the oscillations was a 50 percent insertion into the top of the core of a 1.5 percent reactivity CEA bank for 1 hour. The damping factor for this case was calculated to be about +0.02 per hour; however, when corrected for finite time mesh intervals by the methods of Reference 3.4-12, the damping factor is increased to approximately +0.04. When this damping factor is plotted at the appropriate eigenvalue separation for this mode at end-of-cycle, it is apparent that good agreement is obtained with the modified Randall-St. John prediction. (99-1)

Calculations performed with both Doppler and moderator reactivity feedback have resulted in damping factors which are essentially the same as those obtained with Doppler feedback alone. This result suggests that the constant power condition which applies to the axial oscillations results in a very weak moderator feedback since the moderator density distribution is fixed at the top and bottom of the core and only the density distribution in between can change.

For the calculated Doppler coefficient of -1.36×10^{-3} $\delta\rho$/(kW/ft), the damping factor toward the end of the burnup cycle is positive. Thus, within the uncertainties in predicting power coefficients and uncertainties in the analyses, there is a prediction of unstable axial xenon oscillations in the absence of any control action. These oscillations are sufficiently slow (the period of oscillation being 25-30 hours) so that there would be sufficient time to control the oscillations. In addition, automatic protection is provided if operator action is not taken to remedy the situation. Regulating Bank 7 CEA's are utilized for controlling axial xenon oscillations.

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DRASTIC dynamic response predictions. An experiment and accompanying analysis were performed by the Aerospace Corp. (Reference 3A-23) to verify the ability of the code to account for a complex geometry shell of revolution subjected to transient asymmetric loads. Loads were applied by means of well-defined explosive charges. Based upon the results of dynamic strain measurements made on the test structure, it is evident that the SABOR 5 - DRASTIC code is capable of solving complex dynamic shell structure problems successfully.

In developing the above finite-element computer codes, (i.e., ASHSD, SAMMSOR-DYNASOR, SABOR 5 - DRASTIC) the authors have independently verified their codes with respect to the results of other established structural programs, classical solutions and as possible to experimental data. The correlations demonstrate that the above programs are capable of solving complex dynamic shell structure problems successfully and that the finite-element method of modeling provides accurate representation of the structural phenomena. The SABOR 5 - DRASTIC code, which has had extensive and successful analytical and experimental correlation (Reference 3A-6) for transient (explosive) asymmetric loading, was used to analyze a core support barrel structure with short-term loading. The results of this well-verified program are identical to these of the finite-element codes ASHSD and SAMMSOR-DYNASOR (which are used in the LOCA analysis) for the same core support barrel problem, demonstrating the ability of these programs to adequately represent and evaluate the effect of a transient load on an axisymmetric structure like the core support barrel.

3A.4 Effects of Thermal Shield Removal

Following the discovery of the thermal shield support degradation at the end of Cycle 5 in July, 1983, the thermal shield was removed. A detailed inspection of the core barrel revealed damage at two thermal shield support lug locations. Repairs to the core barrel comprised of drilling crack arrestor holes at the ends of through-wall cracks and removal by machining of non through-wall cracks.

Analytical evaluations and assessments were performed to demonstrate continued structural adequacy of the reactor internals without the thermal shield for all design loading conditions. Special attention was paid to the core barrel to justify the repairs. A description of the repairs to the core barrel, analyses, and significant results is given in Reference 3A-24.

In conclusion, there was no significant change in the loads and the stresses in the internal structures remained within the ASME Code allowables.

3A.5 Leak-Before-Break Analysis

Leak-Before-Break (LBB) analyses for the reactor coolant system (RCS) main coolant loops, for the pressurizer surge line, and unisolable RCS portions of the safety injection and shutdown cooling piping, which demonstrated that the probability of fluid system piping rupture was extremely low, was reviewed and approved by the commission. (See References 3A-25 through 3A-27.) Accordingly, pursuant to revised GDC 4, the dynamic effects associated with pipe ruptures in the above piping segments, including the effects of pipe whipping and discharging fluids have been 99-71 99-71

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TABLE 3.5-3

UNCERTAINTY SOURCES FOR DNBR CALCULATIONS[e]

Uncertainty Source	Uncertainty Value[f]
Power Measurement	Reactor power dependant[a]
ASI:	
Measurement Drift	Reactor power dependent[b]
Rod Shadowing	±0.015 ASI units
Shape Annealing	±0.015 ASI units
Excore Detector Decalib.	±0.01 ASI units
Bias in ASI Uncertainty	±0.005 ASI units
Power Peaking (F_r^T)[c]	±8.5%
Flow Coastdown[d]	+ 0.13/-0.11 second
Pressure	+ 14/-25 psi
Engineering Factor	±3%
Low Flow Trip	±2% of Tech Spec min flow 99-70
Scram Insertion Time	-0.19/+0.22 seconds
Rod Worth	±10%
Flow Measurement	±4% of Tech Spec min flow
RCS Inlet Temperature	±2.25°F

[a] A portion of the power measurement uncertainty is constant, and a portion depends on reactor power. See Reference 3.5-9, Table 4.3, for the constant value and the Figure giving the power dependent portion.

[b] A portion of the ASI measurement drift uncertainty is constant, and a portion depends on reactor power. See Reference 3.5-9, Table 4.3, for the constant value and the Figure giving the power dependent portion.

[c] Radial Peaking of 6% has been removed from this Table because it is not used in the DNBR calculations.

[d] Time for flow to drop from 90% to 80% of Technical Specification minimum flow.

[e] Reference 3.5-9 except for ASI/Rod Shadowing/Shape Annealing/Excore Detector Decalib. (Reference 3.5-19), Bias in ASI Uncertainty (Reference 3.5-20), and Rod Worth (Reference 3.5-21).

[f] Assumed to be $\pm 2\sigma$.

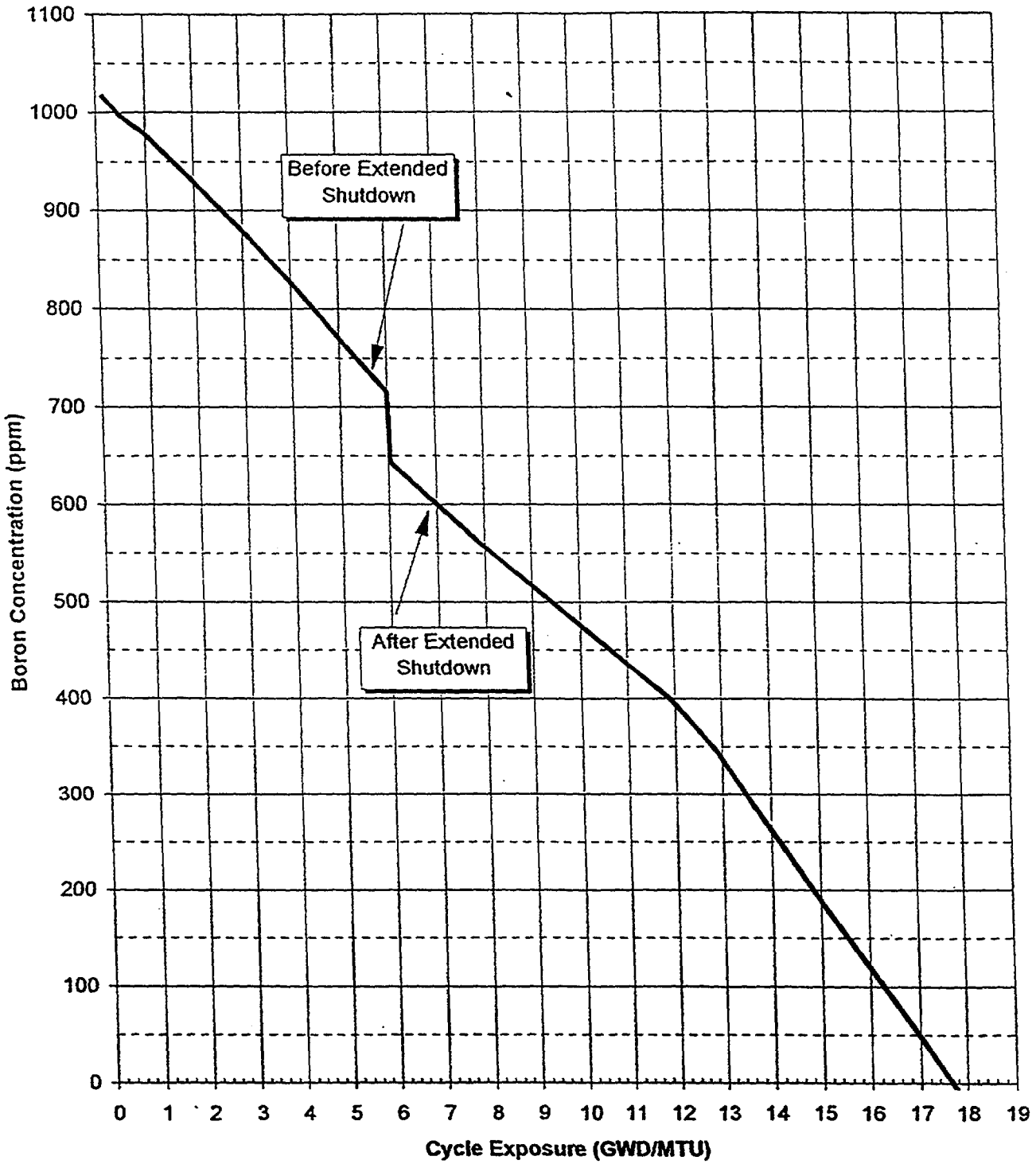

FIGURE 3.4-4
CYCLE 13, BORON LETDOWN, HFP, ARO

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*Note: These Figures are filed in Attachment A to Appendix 5.D.

those specified on the drawings which are supplied by the manufacturers of the equipment installed within the containment.

5.2.3.1.2 Live Loads

Live loads in the containment include design floor loads, equipment live loads, and all loads transmitted through the supports of the enclosure building. A snow load of 60 psf is used for the roof of the enclosure building.

The interior floors and slabs have the following live loads:

a. Floor grating	250 psf
b. Concrete floor slabs	1000 psf
c. Equipment live loads	As specified on drawings supplied by the manufacturers of the various pieces of equipment

5.2.3.1.3 Loss-of-Coolant Accident Loads

The design pressure and temperature of the containment are greater than the peak pressure and temperature that would result from a postulated complete blowdown of the reactor coolant. This could cover a rupture of the reactor coolant system up to and including the severance of the largest reactor coolant pipe.

The supports for the reactor coolant system are designed to withstand the blowdown forces and to restrict the structural deformations associated with the sudden severance of the reactor coolant piping.

Transient pressures and corresponding temperatures resulting from a LOCA or main steam line break accident are presented in Section 14. These serve as the basis for a containment design pressure of 54 psig. 99-73

The variations of temperature with time and the forces resulting from the expansion of the liner plate with the temperature associated with a LOCA are considered.

5.2.3.1.4 Thermal Loads

The spaces between the containment and the enclosure building are maintained at a minimum temperature between 55 and 70°F by unit heaters, as discussed in Section 9.9.2. Under normal operating conditions, a temperature gradient exists when the exterior structure of the concrete cylindrical wall is at 55 to 70°F, while the interior surface is at an operating ambient temperature of 120°F.

However, to be conservative, a design temperature of 20°F (average minimum temperature at the site) is applied at the exterior surfaces of the concrete cylindrical wall in analyzing the temperature gradient under the normal operating conditions.

The temperature gradients through the cylindrical wall of the containment during normal operating and LOCA conditions are shown in Figure 5.2-3.

a. A single break in any pipe under consideration is assumed to occur at one time.

b. The maximum area of the break is assumed to be an opening in the pipe equivalent to its cross-sectional area.

c. The maximum force is based on a free discharge from an open-ended pipe.

The minimum design pressure and temperature of the interior structures are equal to, respectively, the peak pressure and temperature occurring as a result of the complete blowdown of the reactor coolant due to a rupture of the reactor coolant system. This could be up to and including the hypothetical double-ended rupture of the largest pipe in the primary coolant system, with a maximum pipe break size of 19.2 square feet and a peak differential temperature across the primary shield wall of 280°F.

The supports for the reactor coolant system are designed to withstand the blowdown forces associated with the sudden severance of a reactor coolant piping. The 42-inch ID reactor coolant pipe is restrained at the primary shield walls to limit the maximum possible pipe opening to less than 9.8 square feet, which is the cross-sectional area of a double-ended rupture of ID pipe rupture is considered as the maximum design condition for the primary shield walls.

Transient pressures and temperatures resulting from the design base accident and other, lesser accidents are presented in Section 14.6. These serve as the basis for a peak design pressure of 247 psig for the primary shield walls and 7 psig for the secondary shield walls.

The following effects associated with a LOCA are considered:

a. Thrust loads resulting from the rapid mass release from a pipe break in the reactor system or other systems, if the break occurs during a design incident.

b. Pressure build-up in locally confined areas such as the primary shielding cavity or the secondary shielding room.

c. Jet forces resulting from the impingement of the escaping mass upon the adjacent structure.

d. Pipe whipping following a pipe break in the reactor coolant system.

e. Rapid rise in ambient temperature and accompanying rise in ambient pressure.

f. Missiles as described in Subsection 5.2.6.1.1.

With NRC approval of the Leak-before-break (LBB) analyses for the reactor coolant system (RCS) main coolant loops, for the pressurizer surge line, and unisolable RCS portions of the safety injection and shutdown cooling piping, the dynamic effects associated with pipe ruptures in the above piping segments, including the effects of pipe whipping and discharging fluids can be excluded from the design basis. 99-71

functional structure is utilized as a barrier, the loading condition is as determined in either Subsection 5.2.3 for the Containment, Subsection 5.2.5 for the Containment Internals, and Subsection 5.4.3 for the Auxiliary Building. 99-77

5.2.6.8 Pedestal Crane for Containment

(The pedestal crane is no longer in use and has been removed from the containment structure.)

5.2.7 Containment Penetrations

5.2.7.1 Types of Penetrations

All penetrations are pressure resistant, leak-tight, welded assemblies, which are fabricated, installed, inspected, and tested in accordance with the ASME Nuclear Vessel Code, Section III, 1971 and the ANSI Nuclear Piping Code B 31.7.

5.2.7.1.1 Electrical Penetrations

The electrical penetration assemblies through the wall of the containment structure form part of the containment pressure boundary and carry the (NPT) stamp in accordance with ASME Section III (1971). The low voltage power and control modules are mounted in a stainless steel header plate and are designed to meet or exceed all requirements of IEEE Standard 317, 1976. The medium voltage electrical penetration assemblies are double epoxy sealed modules mounted in a stainless steel binder plate and are designed to meet or exceed all requirements pursuant to IEEE 317, 1971, "Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations." In addition to complete electrical prototype tests as described in Section 8.7.2.2, prototype tests are performed which include leakage integrity of 1×10^{-6} std. cc/sec of dry helium under post-accident environmental conditions, seismic integrity, thermocycling, and simulated field installation.

Electrical aspects of these penetrations are described in Section 8.7.2.2.

5.2.7.1.2 Piping Penetrations

Single barrier piping penetrations are provided for all piping passing through the containment walls. Carbon steel pipe sleeves of adequate diameter to allow passage of pipe and insulation where required are supplied.

The closure, forming the single barrier, consists of modified pipe caps, double-flued heads, or closure plates welded to the wall sleeves. The applicable closure types are shown on Figure 5.2-8.

The design data for the containment wall sleeves are included in Subsection 5.2.4.

5.2.7.1.3 Equipment Hatch and Personnel Lock

An equipment hatch, 19 feet in diameter, is provided to permit the transfer of equipment up to and including the size of the reactor vessel head, into and out of the containment. It is fitted with a double-gasketed flange around the dished door to minimize leakage in the unlikely event of a LOCA. Typical details of the equipment hatch are shown in Figure 5.2-4 and 5.2-10.

In addition to the (20) internal bolting assemblies used to secure the equipment hatch during normal operating modes, the equipment hatch has also been outfitted with four external swing bolt attachments which are welded to the equipment hatch ring flange and hatch barrel liner plate. The external attachments provide a method of securing the hatch from the outside during nonpower operation.

A personnel lock is also provided for access into and out of the containment. The personnel lock is equipped with double doors and a quick-acting type, equalizing valve which connects the lock with the interior and exterior of the containment to equalize the pressure in the two systems when personnel are either entering or leaving the containment.

The two doors in the personnel lock are interlocked to prevent both being opened simultaneously, and to ensure that one door is completely closed before the opposite door can be opened. Remote indicating lights and annunciators in the control room indicate the operational status of the door. Provision is made to bypass the interlock system and leave the doors open during plant cold shutdown. The lock interior is provided with lighting and a communication system, each operating from an external supply.

The doors on the Personnel Lock and Equipment Hatch are provided with double gaskets along the closure surfaces. Using the pressure taps furnished, the air space between these two gaskets may be pressurized and checked to assure leak-tightness in accordance with the technical specifications. 99-74

In addition, the Personnel Lock has been designed to withstand the pressurization of the lock chamber to the pressure of 54 psig. During the test, the door on the containment side of the chamber is held closed by a special yoke installed only for the test. This pressurization of the chamber assures the leak-tightness of the door penetrations and the outside lock door under the peak pressure conditions.

Typical details of the personnel lock are shown on Figures 5.2-5 and 5.2-10.

5.2.7.1.4 Fuel Transfer Tube

A fuel transfer tube is provided for fuel movement between the refueling canal in the containment and the transfer canal in the auxiliary building.

The penetration consists of a 36-inch diameter stainless steel tube installed inside a 42-inch sleeve. The transfer tube is fitted with a double-gasketed blind flange in the refueling pool and a standard gate valve in the transfer canal. This arrangement prevents leakage through the transfer tube in the event of an accident. The 42-inch

directly to the containment, such as the steam generator bottom blowdown and demineralized water, are also automatically isolated by the CIAS.

Process lines which are essential for post-accident operation, such as the safety injection, are automatically opened by the Safety Injection Actuation Signal (SIAS). The only lines that automatically open on Containment Spray Actuation Signal are the containment spray headers. The only lines that automatically open on Sump Recirculation Actuation Signal are the containment sump recirculation lines.

5.2.8.4 Availability and Reliability

5.2.8.4.1 Special Features

All containment isolation valves are designed to ensure leak-tightness and reliability of operation. Gate, globe, ball and check valves used for containment isolation meet the leak-tightness requirements of MSS-SP-61 except that the maximum seat leakage rate is less than 2.0 cc/hr. per inch of seat diameter. Butterfly valves used for isolation are tested to ensure essentially zero seat leakage under the test pressure conditions. Subsequent to installation, leak tightness is confirmed by testing as described in Section 5.2.8.1.2.b. The required valve closing times are achieved by appropriate selection of the valve operator type and size.

Containment isolation valve operators that receive the automatic CIAS, and are of the piston (air cylinder) or diaphragm operator type are selected based on the design of the operators being capable of meeting a closure time of 5 seconds. The MSIVs (2-MS-64A and 64B) closure times are described in Section 10.3.2. 99-74

The four containment air purge supply and exhaust valves (2-AC-4, -5, -6 & -7) do not receive a CIAS signal but receive a signal to close in response to a high containment radiation when the plant is in modes 5 and 6 and must meet a 7.5 second closure time. These valves are shut, de-energized by pulling their control power fuse and locked closed by locking their associated fuse block when the plant is in modes 1 through 4. By locking them closed in this manner, these valves are considered sealed closed isolation valves. 99-74

Motor-operated containment isolation valves have no closing stroke time requirements assumed in the accident analysis. The containment spray header and LPSI Injection Header motor-operated isolation valves have opening stroke time requirements that support the system response times assumed in the accident analysis. Refer to the Technical Requirements Manual for the listed system response times. Stroke time requirements are established for motor-operated containment isolation valves in accordance with the Inservice Test Program to ensure the valve operation is monitored for degradation.

Containment penetrations which form closed systems (as defined in GDC 57 inside containment) are not exposed to the containment environment and, therefore, do not constitute a potential leakage path.

The containment penetrations which are open to the atmosphere are listed in Section 5.3.4. For those connected piping systems which will withstand and contain

the post-accident atmosphere outside containment, the closure time of the associated isolation valves is not dictated by potential containment leakage to the site boundary. Any leakage from the Containment Air Purge, Hydrogen Purge, Hydrogen Sample, and Containment Air Monitoring lines is contained within the Enclosure Building Filtration System as stated in Section 5.3.4. (99-55)

The closure time for the containment purge valves is based on 5 seconds. This was demonstrated by testing by the valve manufacturer.

The position of greatest safety for air-operated valves is the position the valve will assume upon loss of air instrument supply.

Motor-operated valves on the shutdown cooling line fail "as is" in the event of loss of electrical power supply. Since these valves are closed during operation (reactor coolant system pressure greater than 300 psig), the "as is" position is the safest position.

In the event of electrical power loss to the motor-operated containment isolation valves, the valves are supplied by emergency power to achieve the greatest safety position.

The assigned locations of containment penetrations are designed to ensure adequate separation of redundant piping and valving. The penetrations leaving the containment below grade are located in the penetration rooms of the auxiliary building while the penetrations above grade are located in the enclosure building. The applicable portions of these structures are designed to protect internal equipment from potential tornadoes and missiles.

Most remotely operated containment isolation valves have provisions for remote manual operation from the main control room. Some of the motor-operated valves have their MCC opening and/or closing coils removed to prevent spurious movement during specific operational modes as shown and detailed in Table 5.2-11. Position indicators (open or closed) are also provided in the control room to assure valve position during an emergency.

The isolation valves which are not required to maintain their full operational capabilities during and after a LOCA or Seismic event are designed to fail in the safe position. These valves are provided with air operators and a spring return to the safe position. The reliability of these valves is proven by seismic analysis and/or testing of the valve, operator, solenoid, and limit switches under the seismic loadings described in Section 5.8. To assure that the valves will operate under the system flow conditions, the air operator and spring are sized to operate at maximum differential pressure across the valve. (99-55)

The isolation valves in vital service which must be capable of full operation during and after a LOCA are provided with motor operators or air accumulator tanks. Electrical power is supplied to the motor-operated valves from the emergency diesel generators during an onsite and/or offsite power-loss to assure that these valves are always capable of full operation. The reliability of the motor-operated valves to function during the seismic disturbance or LOCA is proven by seismic analysis and testing. To demonstrate that the motor operator has sufficient torque to operate the valves, the valves are open and/or closed under full disc differential pressure.

Manually-operated containment isolation valves are positioned using approved operating procedures, a portion of which is a Valve Check List which states the required position(s) of the valve for various plant conditions such as startup or shutdown. As appropriate, the list indicates if the valve is locked open or locked closed. The position of manually-operated containment isolation valves is verified using appropriate valve list(s) before the plant leaves the cold shutdown condition.

The isolation valves required for essential post-accident processes having air operators are provided with emergency reserve air supply tanks which are capable of actuating the valves several times. This is necessary because of the postulated failure of the station and instrument air system during an emergency.

Upon receiving a manual CIAS or SIAS (either manual or automatic) actuation signal, isolation valves required to isolate the containment from the surrounding environment and other systems within the station, close automatically. Valve operators are sized to close these isolation valves before any significant amount of radioactivity can be released from the containment. In most systems, standard valve operators are sufficient.

The containment isolation valve operators have a certified proven record of reliability under operating conditions similar or more severe than those to which exposed during unit operation. These have been tested by the manufacturer to ensure the integrity in the event of inadvertent closure under operating conditions.

The worst environment to which containment isolation valves may be subjected is that described in Section 14.8.2 Containment Analysis. In addition, static, dynamic, and seismic loads, as described in Section 5.8, and exposure to physical damage are taken into account. Valves have been designed to perform their intended function under these conditions and this forms their design basis. (99-55)

Damage due to severe natural environment such as freezing is not considered credible since all areas which house containment isolation valves are maintained at temperatures above freezing.

Relief valves are required to prevent over-pressuring lines for low-pressure service but subject to possible valve leakage from lines transporting fluids of potentially high temperature and pressure. Relief valves are also required to prevent over-pressuring lines due to uncontrolled thermal expansion of the process fluid.

Penetration No. 10, Figure 5.2-29, transports reactor coolant during shutdown cooling, normally at 300 psig, but it is also subject to the maximum pressure of reactor coolant system during normal plant operation. To prevent over-pressuring the piping external to the containment due to isolation valve leakage, a relief valve set at 300 psig is required. This relief valve is inside the containment and discharges to the liquid radwaste system, as shown in Figure 6.1-1. (99-55)

Penetration No. 11, Figure 5.2-31, is for the Safety Injection tank testing and RCS check valve leakage bleed. The relief valve is set at 450 psig to protect the piping from over-pressure. This relief valve is also inside the containment and discharges to the Quench Tank, as shown in Figure 6.1-1. (99-55)

Penetration Nos. 19 & 20, Figure 5.2-33, are for the main steam lines with relief valves required to protect the steam generator from over-pressure. There are 8 relief valves per main steam line for a total of 16 relief valves with nominal set points at 985 psig and the maximum pressure settings at 1,035 psig. A detailed description of these valves is given in Section 4.3, Table 4.3-3 of the FSAR. The relief valves are located outside the containment to facilitate inspection, testing, and maintenance and to protect the containment from over-pressure due to relief discharge. (99-55)

Penetration Nos. 12 & 13, Figure 5.2-30, provide containment sump recirculation for long-term operation of ECCS and containment spray post accident. Piping equipped with rupture disks are connected to the body drains of containment isolation valves 2-CS-16.1A & B. The rupture disks discharge is contained within the closed piping system. The rupture disks prevent the possibility of the motor-operated valves becoming pressure locked in the closed position due to thermal expansion of trapped fluid in the valve bodies prior to initiation of sump recirculation.

5.2.8.4.2 Tests and Inspections

All isolation valves are shop tested and examined by the manufacturer in accordance with the governing code requirements to assure the integrity of the pressure-retaining boundary. In addition, all valves are performance tested for seat leakage on an individual valve basis to assure reliability.

Each valve is tested after installation to ensure its leak-tightness and performance. The valve operators specified have a proven record of a number of years of reliability in respect to the method of operation and material used. Throughout the plant life, these valves are tested periodically as required per 10 CFR Part 50, Appendix J and those which cannot be tested during operation (those which must remain open or closed) are tested during the scheduled shutdowns and plant outages.

Containment fluid penetration isolation valves are incorporated with provisions for periodic leak rate testing provided they are required to be tested per 10 CFR Part 50, Appendix J.

The containment isolation valves located outside the containment are accessible for maintenance and inspection during normal plant operation. The isolation valves located within containment are accessible during normal plant shutdown for maintenance and inspection.

On-line testing procedures and preoperation or acceptance testing of isolation valves are discussed with the respective process systems in Chapters 6, 9, and 10. (99-55)

5.2.9 Containment Testing And Surveillance

5.2.9.1 Integrated Leak-Rate Surveillance Test Program

A containment test program has been established to assure reactor containment building will adequately protect the public from core damage accidents and achieve compliance with 10CFR50, Appendix J, of the Code of Federal Regulations.

Containment leak tests are performed periodically per the requirements of Appendix J.

The objective of these tests is to demonstrate that leakage through the primary reactor containment and systems, and components penetrating the primary containment, do not exceed the allowable leakage rate specified in the Plant Technical Specifications (less than 0.75 L_a). (97-213)

The containment penetrations are aligned in a post-LOCA configuration (i.e., plant systems penetrating the containment boundary isolated, via normal closure modes, drained of water and vented inside and outside of isolation valves) to the extent practical. A pressurization system is set up and connected to the containment through a temporary piping path. The pressurization system consists of a group of oil-free air compressors, dryer units, after-coolers, interconnecting spool pieces, and valves. (95-19)

Refer to Table 5.2-14 for details of the leak-rate measurement system instrumentation.

A data acquisition system (with backup capability) is used to record ILRT containment-related test parameters, e.g., containment air pressure, temperatures, and dew point temperatures. The data acquisition system typically consists of a portable computer, a data logger, and a printer. The test data is processed via a quality-related ILRT computer program. (95-19)

When test prerequisites and initial conditions are satisfied, the containment is pressurized to slightly greater than accident pressure with external leak checks performed to identify any containment leakage. When test pressure is reached, containment pressurization is stopped and isolated. The containment air mass system is then allowed to thermodynamically stabilize. Once stabilization is attained, the data acquisition system records the test data and computes the ILRT leakage rate, L_{am}. (95-19) (97-213)

The Type A test and the supplemental verification test are performed according to the requirements of the MP2 Technical Specification and 10CFR50, Appendix J.

Two methods are available to calculate the containment leakage rate; mass point and total time.

The mass point method uses formulas from ANS 56.8 (Ref. 5.2-8) and the total time method from BN-TOP-1 (Ref. 5.2-9)

Quality-related software employing these techniques calculates the least-squares fit and upper confidence limit containment leakage rate, Lam. It automatically checks it against the acceptance criteria (0.75 La). (95-19)

The software program containment model inputs are based upon quality-related engineering calculations: containment-free air volume V, Resistance Temperature Detector (RTD), and Dewcell sensor volume weight fractions (refer to Table 5.2-15 and Ref. 5.2-12 for details), and the superimposed leakage rate, L_o. (95-19)

A single failure RTD and Dewcell analysis calculation is completed and reweighted plan is established, per the guidance of EPRI (Ref. 5.2-10). (95-19)

Prior to depressurization of the containment, a verification test is completed. The verification test induces a known leakage rate, Lo, and a composite leakage calculation of L_c is made to verify that the test instrument data acquisition system was operating satisfactorily ($L_{o+} L_{am} - 0.25 L_a \leq L_c \leq L_o + L_{am} + 0.25 L_a$) and yielding accurate results.

Once this is verified, the containment is then slowly depressurized to normal atmospheric conditions and restoration is started.

5.2.9.1.1 Total Time Method for Calculating Containment Leakage Rate (95-14)

The Total Time Method of the Absolute Method consists of calculating air lost from the containment using pressure, temperature, and dew point observations made during the ILRT using the Ideal Gas Law. The measured leakage rate at any time (t) is determined by calculating the percent leakage rate based on the most recent data and the data taken at the start of the test. The calculated leakage rate is then determined by plotting the measured leakage rate as a function of time and then performing a least-squares fit of the measured leakage rate values. The calculated leakage rate is expressed as a percentage of containment mass lost in a 24-hour period. (97-200)

This is the primary calculation (data analysis) for use during a short duration test (i.e., test duration less than 24 hours). Bechtel Topical Report BN-TOP-1, Rev. 1, 1972, "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants" (Reference 5.2-9) and ANSI N45.4-1972, Leakage Rate Testing of Containment Structures for Nuclear Reactors" (Reference 5.2-15), provide details on this method of determining the containment leakage rate.

5.2.9.1.2 Mass Point Method for Calculating Containment Leakage Rate

The Mass Point analysis technique consists of calculating the air mass within the containment structure over the test period using pressure, temperature, and dew point observations made during the ILRT using the Ideal Gas Law. The leakage rate is then determined by plotting the air mass as a function of time using a least-squares fit to determine the slope. The leakage rate is expressed as a percentage of air mass in a 24-hour period. This is the primary calculation (data analysis) for test durations of 24 hours or greater. This analysis technique is described in ANSI/ANS 56.8, "Containment System Leakage Testing Requirements" (Reference 5.2-8). (97-200)

5.2.9.2 Structural Integrity Test

Prior to operation the containment was subjected to a pressure test equivalent to 115 percent of the postulated maximum accident pressure in accordance with Regulatory Guide 1.18. This test provides a direct verification of the structural integrity of the containment as a whole, i.e., it is equal to or better than that required to sustain the forces imposed by the accident loading condition.

The structure was pressurized to 115 percent of the postulated accident pressure (62.4 psig) during the integrated leak rate test. Radial measurements were made along six equally spaced meridians at locations near the base, at mid-height of the cylinder,

and at the spring line as well as the top of the dome. Radial deflections were also taken at 12 positions around the equipment hatch.

Cracks greater than 0.005 inch were mapped near the base wall intersection, at mid-height, just below the ring girder, at the intersection of a buttress and the wall, and at the dome. At each location an area of at least 40 square feet was mapped.

Measurements and observations were taken at 0, 14, 27, 40, 54, and 62.4 psi increments while pressurizing and depressurizing. The results of these provided de facto indication that the structure is more than adequate to withstand the design internal pressures.

5.2.9.3 Post-Operational Leakage Monitoring

The procedure deviates from the Regulatory Guide 1.18 in that the tangential deflections at 12 positions around the equipment hatch were not measured since tangential deflections were not predicted. Tangential deflections measured during test on similar structure were of minimal significance.

5.2.9.4 Tendon Surveillance

5.2.9.4.1 Program Description

The primary objective of the surveillance program for the containment structure concrete and tendons during the lifetime of the plant is to ensure the strength and reliability of the post-tensioning steel and other major components such as stressing washers, shims, and bearing plates. The surveillance program is intended to provide sufficient in-service historical evidence to maintain a high level of confidence so that the integrity of the containment structure may be preserved.

This program consists of the following operations:

a. Recording lift-off pressure readings and any significant visual difference of stressing washers, shims, bearing plates, and concrete cracks.

b. Checking for possible corrosion of wires and anchorage components.

To accomplish this surveillance program, a total of twenty-one (21) tendons were provided in accordance with the Regulatory Guide 1.35, Revision 1 as follows:

a. Horizontal (hoop): Ten (10) tendons randomly selected but approximately equally distributed.

b. Vertical: Five (5) tendons are located in the wall, approximately equally spaced around the containment.

c. Dome: Six (6) tendons, two located in each 60 degrees group.

d. For the fourth surveillance (tenth year), if no significant problems were indicated by the previous surveillances, the total number of surveillance tendons could be reduced to three from each group for a total of nine surveillance tendons.

In July 1990, Regulatory Guide 1.35, Revision 3, was published. Revision 3 requires a random selection from all tendon groups. Under Revision 3, the third surveillance would have required a total of fourteen surveillance tendons to be inspected, while only nine would be required for the fourth surveillance onward assuming no problems were found in the earlier surveillances. Accordingly, from the fifth surveillance on the tendon selection, inspections, testing and sampling is being performed in accordance with Regulatory Guide 1.35, Revision 3. The sampling may be expanded for any surveillance when it is deemed desirable or necessary.

5.2.9.4.2 Compliance with Regulatory Guide

The tendon surveillance program complies with the Regulatory Guide 1.35.

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5.2-7 10CFR50, Appendix J, Primary Reactor Containment Leakage Testing for Water-Cooled Power Reactors.

5.2-8 ANS 56.8, "Containment System Leakage Testing Requirements." (95-19)

5.2-9 BN-TOP-1, Revision 1, November 1, 1972. Bechtel Corporation Testing Criteria for Integrated Leak-Rate Tests of Primary Containment Structures for Nuclear Power Plants.

5.2-10 EPRI Report NP-2726, "Containment Integrated Leak-Rate Testing Improvements," November 1982.

5.2-11 T2621-P, MP2 Preoperational Test, April 15, 1975.

5.2-12 Engineering Calculation No. 95-ENG-1184-M2, "MP2 Containment ILRT Sensor Volume Fractions." (95-19)

5.2-13 Millstone Unit 3, Final Safety Analysis Report, Section 2.3-Meteorology.

5.2-14 Richard Jr., F. E., Hall Jr., J. R., and Woods, R. E., Vibrations of Soils and Foundations, Prentice Hall, Inc., NJ, 1970. (97-213)

5.2-15 ANSI N45.4-1972, "Leakage Rate Testing of Containment Structures for Nuclear Reactors." (97-200)

5.3 ENCLOSURE BUILDING

5.3.1 General Description

The enclosure building is a limited leakage steel framed structure with uninsulated metal siding and an insulated roof deck. It also includes those portions of the auxiliary building adjacent to the containment, as shown in Figures 5.3-1 and 5.3-2. The enclosure building completely surrounds the containment above grade and is designed and constructed to ensure that an acceptable upper limit of leakage of radioactive materials to the environment would not be exceeded in the unlikely event of a loss-of-coolant incident.

The use of a suitable formed gasket material at all joints will provide assurance that the required degree of air-tightness and partial vacuum will be maintained within the enclosure building. Two continuous lines of caulking are provided at all lap joints of both siding and decking, with the exception of the east and west wall blowout panels. A single line of caulking shall be provided at the bottom end lap and side laps of the blowout panels. The caulking may be applied to either the interior building seam or the exterior building seam. 98-86 99-20

Principal dimensions of the enclosure building are as follows:

Length (ft)	153.0
Width (ft)	147.0
Height (ft)	147.0
Decking (gauge)	20
Siding (gauge)	22

The enclosure building is supported partially on concrete grade beams and caissons, partially on the roof of the auxiliary and turbine buildings, and partially on the dome of the containment. The interior of the enclosure building contains permanent ladders, stairways and catwalks which provide access to the upper exterior regions of the containment and to equipment in this building. In addition, permanent work platforms are furnished for the periodic surveillance of the post-tensioned prestressing tendons.

Concrete floor slabs are provided at grade between the enclosure building and the containment, and also at Elevations 36-6 and 38-6. A waterproofing membrane is provided under the slabs at grade and is extended down and around the containment below grade, as shown on Figure 5.3-3. Between the waterproofing membrane and the containment wall, corrugated asbestos-cement siding is installed as shown in Figure 5.3-4, to provide a passage of least resistance for possible leakage from the containment below grade to the enclosure building.

There are two stacks that exhaust radioactive effluents from the Millstone Unit 2 operations. Radioactive effluents are piped to the ventilation stack that was provided for the Millstone Nuclear Power Station, Unit 1. This stack provided for future expansion to accept effluent gases from the Unit 2 plant. The physical features of the stack are provided in Section 3.8 of the FSAR for the Millstone Nuclear Power Station, Unit 1. Gas volume increase is less than one percent, resulting in an exit gas velocity of 5,723 feet per minute. Section 3.8, of the Unit 1, FSAR lists the ventilation stack as a Class I structure. This section outlines the design criteria for earthquake loading and the dynamic analysis applied to the structure. Stack failure would not directly impact any safety related equipment.

The only other stack that exhausts radioactive effluents to the atmosphere from Unit 2 is the stack located atop the enclosure building. The stack is constructed of 1/4 inch steel plate and standard structural shapes. Overall height is 13 feet. This is a seismic Class I stack, designed in accordance with the criteria contained in Section 5.3.3 of the FSAR. The stack has a constant rectangular cross section which has dimensions of 4'0" x 9'6". Exit velocity of effluents is 1,684 feet per minute with two fans operating and 2,526 feet per minute with three fans operating. During normal plant operation, two fans are operating. (2/89)

5.3.2 Construction Materials

The following materials are used in the construction of the enclosure building:

a. Structural steel	ASTM A-36
b. Concrete (psi)	
Grade beams and Caissons	4,000
Slabs at grade	3,000
Floor slabs	3,000
c. Reinforcing steel	ASTM A615, Grade 60
d. Metal siding	22 gauge
e. Metal roof decks	20 gauge

5.3.3 Design Bases

The design of the enclosure building provides the required features as outlined in General Design Criteria 1, 2, 3, 4, 5, 60, Appendix A of 10 CFR Part 50. (2/89)

5.3.3.1 Bases for Design Loads

The following loads are considered in the design of the enclosure building:

a. Dead loads
b. Live loads including external pressures
c. Earthquake loads
d. Wind and tornado loads

5.3.3.1.1 Dead Loads

The dead loads consist of the weight of the steel frame, roof, metal siding, and access stairs and ladders.

5.3.3.1.2 Live Loads

The design live loads for the enclosure building are as follows:

a. Roof, snow loads (psf)	60	Admin.
b. Slabs at grade		
Equipment hatch area	AASHO H-20 truck load	
Other areas (psf)	500	
External pressure (independent of wind and tornado loads) (psf)	10	

Weights of equipment as indicated on drawings supplied by the manufacturer are included as live loads.

5.3.3.1.3 Earthquake Loads

The earthquake loads are predicated on an operating basis earthquake (OBE) at the site having a horizontal ground surface acceleration of 0.09 g. In addition, a design basis earthquake (DBE) having a horizontal ground surface acceleration of 0.17 g is used to check the design of the enclosure building to ensure no loss of structural function. The seismic design spectrum curves are given in Figure 5.8-1 and 5.8-2. A vertical component two-thirds of the magnitude of the horizontal ground surface component is applied at the base simultaneously.

A dynamic analysis including the effects of the attachments to the other structures is used to arrive at the equivalent static loads for the design. (1/89)

5.3.3.1.4 Wind and Tornado Loads

Winds loads for the design of the enclosure building are based on a wind velocity of 115 mph with gusts up to 140 mph. The ASCE Paper 3269 is used to determine the shape factors. However, the provisions in the paper for gust factors and variations of wind velocity with respect to height are not applied.

The entire enclosure building is designed to resist the effects of the 140 mph wind.

Tornado loads on the enclosure building are based on a tornado funnel having a peripheral tangential velocity of 300 mph and a translational velocity of 60 mph. These velocities are combined, resulting in a design basis tornado wind velocity of 360 mph. The enclosure building, adjacent to structures which house safety related equipment, is designed so that its structural framing will withstand tornado winds, but the siding will be blown away.

The wind velocity is assumed to be uniformly distributed over the height of the structure. Probable missiles in the form of siding are less critical than the design missiles as specified in Section 5.2.6.1.2 of the FSAR. The siding when blown off may induce superficial damage to the adjacent structures, but the structural integrities of the adjacent structures will be maintained. The design requirements for tornado loads for structures which house safety related equipment for shutdown are given in Section 5 of the FSAR.

TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification	Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position	
1	Demineralized Water	PMW	IA	N	IN	1B	Yes	2-PMW-43	Outside	2"-Globe-1	2"	Diaphragm	CIAS	Closed	Closed	Yes	Closed	(99-55
							No	2-PMW-165	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed	
2	Letdown Line To Purification Demin.	CVCS	IA	P	OUT	7	Yes	2-PMW-3	Inside	2"-Check-1		----	----	----	----	No	----	(99-55
							Yes[4]	2-CH-516	Inside	3"-Globe-1	2"	Diaphragm	CIAS	Open	Closed	Yes	Closed	(99-55
							No	2-CH-006	Inside	2"-Gate-1		Manual	----	Open	As Is	No	Open	
							Yes	2-CH-089	Outside	2"-Globe-1		Diaphragm	CIAS	Open	Closed	Yes	Closed	
							No	2-CH-763, 658, 991	Inside	1"-Gate-3		Manual	----	Locked Closed	As Is	No	Closed	(99-27
							No	2-CH-260, 082, 083	Inside	3/4"-Globe-3		Manual	----	Locked Closed	As Is	No	Closed	
							No	2-CH-067	Outside	3/4"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed	
3	Reactor Coolant Charging Line	CVCS	IA	P	IN	9	Yes[4]	2-CH-515	Inside	3"-Globe-1		Diaphragm	SIAS	Open	Closed	Yes	Closed	
							Yes[4]	2-CH-518, 519	Inside	2"-Globe-2	2"	Diaphragm	Remote	Open	Open	Yes	Open	(99-55
							Yes[4]	2-CH-517	Inside	2"-Globe-1		Diaphragm	Remote	Closed	Closed	Yes	Closed	
							No	2-CH-431	Inside	2"Spring-Check-1		------	----	----	----	No	----	
							Yes[4]	2-CH-434	Inside	2"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed	(99-55
							Yes	2-CH-429	Outside	2"-Gate-1		MOV	Remote	Open	As Is	Yes	Open	
							No	2-CH-004, 003	Inside	3/4"-Gate-2		Manual	----	Locked Closed	As Is	No	Closed	
							No	2-CH-001, 002, 443, 698, 697, 714	Inside	3/4"-Globe-6		Manual	----	Locked Closed	As Is	No	Closed	

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

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TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification		Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
							No	2-CH-710		Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							Yes[4]	2-RC-71		Inside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-661		Inside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-752, 753		Inside	1"-Gate-2		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-435		Inside	2"-Spring Check-1[5]		----	----	----	----	No	----
							No	2-CH-986		Outside	3/4"x1"-Relief-1		----	----	----	----	No	----
4	Containment Spray Water	CSS	IA	O	IN	17A	Yes	2-CS-5A		Inside	8"-Check-1	8"	-----	----	----	----	No	----
							Yes	2-CS-4.1A		Outside	8"-Gate-1		MOV	CSAS	Closed	As Is	Yes	Open
							No	2-CS-049C		Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CS-049A		Outside	1"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
5	Containment Spray Water	CSS	IA	O	IN	17B	Yes	2-CS-5B		Inside	8"-Check-1	8"	-----	----	----	----	No	----
							Yes	2-CS-4.1B		Outside	8"-Gate-1		MOV	CSAS	Closed	As Is	Yes	Open
							No	2-CS-101		Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
6,8	Safety Injection Low & High Pressure	SIS	IB	P	IN	10A(Pene. 6)	No	Pene. 6 2-SI-645	Pene. 8 2-SI-635	Outside	6"-Globe-1	6"	MOV	SIAS	Closed	As Is	Yes	Open
						10B(Pene. 8)	No	2-SI-646, 647	2-SI-636, 637	Outside	2"-Globe-2		MOV	SIAS	Throttled	As Is	Yes	Open

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification		Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position	
							No		2-SI-733, 041E	Outside	(Pene. 8 only) 1"-Globe-2		Manual	----	Locked Closed	As Is	No	Closed	99-97
							No	2-SI-144	2-SI-134	Outside	6"-Check-1		----	----	----	----	No	----	
							No	2-SI-143, 009	2-SI-133, 010	Outside	2"-Check-2		----	----	----	----	No	----	
							No	2-SI-095, 734		Outside	(Pene. 6 only) 1"-Globe-2		Manual	----	Locked Closed	As Is	No	Closed	
							No	2-SI-041F, 735, 742D	2-SI-041D, 110, 742C	Outside	3/4"-Globe-3		Manual	----	Locked Closed	As Is	No	Closed	
							No	2-SI-145, 146	2-SI-135, 136	Outside	3/4"-Globe-2		Manual	----	Open	As Is	No	Open	99-97
							No	2-SI-706D	2-SI-706C	Inside	6"-Check-1		----	----	----	----	No	----	99-97
7	Safety Injection Low & High Pressure	SIS	IB	P	IN	10C	No	2-SI-615		Outside	6"-Globe-1	6"	MOV	SIAS	Closed	As Is	Yes	Open	
							No	2-SI-616, 617		Outside	2"-Globe-2		MOV	SIAS	Throttled	As Is	Yes	Open	
							No	2-SI-114		Outside	6"-Check-1		----	----	----	----	No	----	
							No	2-SI-012, 113		Outside	2"-Check-2		----	----	----	----	No	----	
							No	2-SI-041A, 107, 716, 715, 742A		Outside	3/4"-Globe-5		Manual	----	Locked Closed	As Is	No	Closed	
							No	2-SI-717, 718		Outside	1"-Gate-2		Manual	----	Locked Closed	As Is	No	Closed	
							No	2-SI-115, 116		Outside	3/4"-Globe-2		Manual	----	Open	As Is	No	Open	99-97
							No	2-SI-706A		Inside	6"-Check-1		----	----	----	----	No	----	99-97
9	Safety Injection Low & High Pressure	SIS	IB	P	IN	10D	No	2-SI-625		Outside	6"-Globe-1	6"	MOV	SIAS	Closed	As Is	Yes	Open	

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

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TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification		Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
							No	2-SI-626, 627		Outside	2"-Globe-2		MOV	SIAS	Throttled	As Is	Yes	Open
							No	2-SI-124		Outside	6"-Check-1		----	----	----	----	No	----
							No	2-SI-123, 011		Outside	2"-Check-2		----	----	----	----	No	----
							No	2-SI-722, 723, 720, 721, 742B		Outside	3/4"-Globe-5		Manual	----	Locked Closed	As Is	No	Closed
							No	2-SI-125, 126		Outside	3/4"-Globe-2		Manual	----	Open	As Is	No	Open
							No	2-SI-706B		Inside	6"-Check-1			----	----	----	No	----
10	Reactor Coolant Shutdown Cooling	SIS	IB	P	OUT	11	Yes	2-SI-709		Outside	12"-Gate-1	12"	Manual	----	Locked Closed	As Is	No	Closed
							Yes[4]	2-SI-651		Inside	12"-Gate-1		MOV	Remote[7]	Closed	As Is	Yes	Closed
							No	2-SI-101A		Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-SI-102A		Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-SI-043A		Inside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
11	Safety Injection Tank Test Line	SIS	IA	N	OUT	20	Yes	2-SI-463		Outside	2"-Gate-1	2"	Manual	----	Locked Closed	As Is	No	Closed
12 & 13	Containment Sump Recirc. Line	SIS	Special	Special	OUT	16	No	Pene. 12 2-CS-16.1A	Pene. 13 2-CS-16.1B	Outside	24"-Gate-1	24"	MOV	SRAS	Closed	As Is	Yes	Open
							No	2-CS-125	2-CS-127	Outside	3/4"-Gate-1		Manual	----	Locked Open	As Is	No	Open
							No	2-CS-130	2-CS-135	Outside	3/4"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CS-131	2-CS-136	Outside	3/4"-Gate-1		Manual	----	Locked Open	As Is	No	Open
							No	2-CS-132	2-CS-137	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed

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Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

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TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification		Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
							No	2-CS-133	2-CS-138	Outside	3/4"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CS-134	2-CS-139	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CS-140	2-CS-141	Outside	3/4"-Check-1		----	----	Closed	As Is	No	Closed
14	Containment Sump to Aerated Drain Tk.	RWS	IA	O	OUT	13A	Yes	2-SSP-16.2		Outside	3"-Globe-1	3"	Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes	2-SSP-16.1		Inside	3"-Globe-1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							No	2-SSP-51		Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-SSP-73		Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
15 & 16	Feedwater & Aux. Feedwater	FW	II	N	IN	15A	No	Pene. 15 2-FW-5A	Pene. 16 2-FW-5B	Outside	18"-Stopcheck-1	18"	Air Cyl.	MSI	Open	Closed	Yes	Closed
						for Pene. 15	No	2-FW-12A	2-FW-12B	Outside	6"-Stopcheck-1		Air Cyl.	Remote	Closed	Closed	Yes	Closed
						15B for Pene. 16	No	2-FW-16A	2-FW-16B	Outside	1"-Check-1		----	----	----	----	No	----
							No	2-FW-15A	2-FW-15B	Outside	1"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-FW-86	2-FW-182	Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-FW-261A		Outside	(Pene. 15 only) 3/4"-Globe-1		Manual	---	Locked Closed	As Is	No	Closed
19 & 20	Main Steam	MSS	III	N	OUT	23	No	Pene. 19 2-MS-64A	Pene. 20 2-MS-64B	Outside	34"-Stopcheck-1	34"	Air Cyl.	MSI	Open	Closed	Yes	Closed[6]

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification		Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
							No	2-MS-371	2-MS-369	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-MS-201	2-MS-202	Outside	4"-Gate-1		MOV	Remote[7]	Open	As Is	Yes	Open
							No	2-MS-3A	2-MS-3B	Outside	12"-Gate-1		Manual	----	Open	As Is	No	Open
							No	2-MS-190A	2-MS-190B	Outside	8"-Globe-1		Diaphragm	SG Press.	Closed	Closed	Yes	Closed
							No	2-MS-265B	2-MS-266B	Outside	1"-Globe-1		Diaphragm	MSI	Open	Closed	Yes	Closed
							No	2-MS-247, 248, 249, 250, 251, 252, 253, 254	2-MS-239, 240, 241, 242, 243, 244, 245, 246	Outside	6"-Relief-8		----	----	----	----	No	----
							No	2-MS-65A	2-MS-65B	Outside	3"-Globe-1***		MOV	MSI[7]	Closed	As Is	Yes	Closed
							No	2-MS-297	2-MS-296	Outside	1"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-MS-265A	2-MS-266C	Outside	1"-Gate-1		Manual	----	Open	As Is	No	Open
							No	2-MS-255		Outside	(Pene. 19 only) 3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No		2-MS-258	Outside	(Pene. 20 only) 1"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-MS-41A	2-MS-41B	Outside	3/4"-Globe-1		Manual	----	Open	As Is	No	Open
							No	2-MS-461	2-MS-462	Outside	3/4"-Globe-1		Manual	----	Closed	As Is	No	Closed
							No	2-MS-459	2-MS-458	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Locked Closed
21	Reactor Coolant & Pressurizer Sampling	SS	IA	P	OUT	19	No	2-LRR-265		Inside	1/2"-Check-1	1/2"	----	----	----	----	No	----

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification		Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
							Yes	2-LRR-61.1		Inside	1/2"-Globe-1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes	2-RC-45		Outside	3/4"-Globe-1		Diaphragm	CIAS	Open	Closed	Yes	Closed
							Yes	2-RC-001, 002		Inside	3/4"-Globe-2		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes	2-RC-003		Inside	3/4"-Globe-1		Diaphragm	CIAS	Open	Closed	Yes	Closed
							No	2-RC-65		Inside	3/8"-Globe-1		Manual	----	Open	As Is	No	Open
							No	2-RC-434, 435		Inside	3/8"-Globe-2		Manual	----	Locked Closed	As Is	No	Closed
22 & 23	Steam Generator Bottom Blowdown	SGBS	IA	N	OUT	14	Yes	Pene. 22 2-MS-220A	Pene. 23 2-MS-220B	Outside	2"-Globe-1	2"	Diaphragm	AFAIS, CIAS & Hi CTMT Radiation Hi Rad	Open	Closed	Yes	Closed
24	Reactor Bldg. Closed Cooling Water Inlet to Reactor Coolant Pumps and Other Components[8]	RBCCW	IA	N	IN	24	Yes	2-RB-30.1A		Outside	8"-Gate-1	8"	MOV	Remote	Open	As Is	Yes	Open
							No	2-RB-289		Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-RB-173A		Outside	3/4"-Globe-1		Manual	----	Open	As Is	No	Open
25 & 26	Reactor Bldg. Closed Cooling Water to Containment Air Recirculation Units	RBCCW	IA	N	IN	21A	No	Pene. 25 2-RB-28.1D	Pene. 26 2-RB-28.1B	Outside	10"-Butterfly-1	10"	Air Cyl.	Remote	Open	Open	Yes	Open
							No	2-RB-282	2-RB-283	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No		2-RB-345	Outside	(Pene. 26 only) 1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed

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Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification		Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
27 & 28	Reactor Bldg. Closed Cooling Water to Containment Air Recirculation Units	RBCCW	IA	N	IN	21B	No	Pene. 27 2-RB-28.1A	Pene. 28 2-RB-28.1C	Outside	10"-Butterfly-1	10"	Air Cyl.	Remote	Open	Open	Yes	Open
							No	2-RB-236	2-RB-237	Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
29	Reactor Bldg. Closed Cooling Water Outlet from Reactor Coolant Pumps and Other Components[6]	RBCCW	IA	N	OUT	2	Yes	2-RB-37.2A		Outside	8"-Gate-1	8"	MOV	Remote	Open	As Is	Yes	Open
							No	2-RB-297A		Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-RB-298		Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
30, 31, 32 & 33	Reactor Bldg. Closed Cooling Water From Contain. Air. Recirc. Cool.	RBCCW	IA	N	OUT	22	No	Pene. 30 2-RB-28.3D Pene. 31 2-RB-28.3B	Pene. 32 2-RB-28.3A Pene. 33 2-RB-28.3C	Outside	10"-Butterfly-1	10"	Air Cyl.	SIAS	Closed	Open	Yes	Open
							No	Pene. 30 2RB-28.2D Pene. 31 2RB-28.2B	Pene. 32 2RB-28.2A Pene. 33 2RB-28.2C	Outside	6"-Butterfly-1	6"	Air Cyl.	Remote	Open	Open	Yes	Open
34	Nitrogen Supply	NS	IA	N	IN	18	Yes	2-SI-312		Outside	3/4"-Globe-1	1"	Diaphragm	CIAS	Open	Closed	Yes	Closed
							No	2-SI-045		Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
35	Drain from Primary Tank	RWS	IA	O	OUT	13B	Yes	2-LRR-43.2		Outside	3"-Globe-1	4"	Diaphragm	CIAS	Closed	Closed	Yes	Closed
							Yes[4]	2-LRR-43.1		Inside	3"-Globe-1		Diaphragm	CIAS	Closed	Closed	Yes	Closed

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Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

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TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification	Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
							No	2-LRR-291	Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-LRR-293, 295	Outside	3/4"-Globe-2		Manual	----	Locked Closed	As Is	No	Closed
36	Instrument Air	IA	IA	N	IN	33	Yes	2-IA-566	Outside	1/2"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed[9]
							Yes	2-IA-569	Inside	1/2-Check-1		----	----	----	----	No	----
							No	2-IA-572	Inside	1/2"-Gate-1		Manual	----	Locked closed	As Is	No	Closed
37	Instrument Air	IA	IA	N	IN	1A	Yes	2-IA-27.1	Outside	2"-Globe-1	2"	Diaphragm	Remote	Open	Closed	Yes	Open[9]
							No	2-IA-40	Outside	1"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							Yes	2-IA-43	Inside	2"-Check-1		----	----	----	----	No	----
38	Station Air	SA	IA	N	IN	3	Yes	2-SA-19	Outside	2"-Gate-1	2"	Manual	----	Locked Closed	As Is	No	Closed
							No	2-SA-28	Inside	1"-Gate-1	2"	Manual	----	Locked Closed	As Is	No	Closed
							Yes	2-SA-22	Inside	2"-Check-1		----	----	----	----	No	----
39	Purge Air Inlet	PA	IC	O	IN	4	Yes	2-AC-4	Outside	48"-Butterfly-1	48"	Air Cyl.	Hi CTMT Radiation	Locked Closed	Closed	Yes	Closed
							Yes[4]	2-AC-5	Inside	48"-Butterfly-1		Air Cyl.	Hi CTMT Radiation	Locked Closed	Closed	Yes	Closed
							No	2-AC-21	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
40	Purge Air Discharge	PA	IC	O	OUT	5	Yes	2-AC-7	Outside	48"-Butterfly-1	48"	Air Cyl.	Hi CTMT Radiation	Locked Closed	Closed	Yes	Closed
							Yes[4]	2-AC-6	Inside	48"-Butterfly-1		Air Cyl.	Hi CTMT Radiation	Locked Closed	Closed	Yes	Closed

(99-55)

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List." (99-55)

TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification	Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
42	Fuel Transfer Tube	FTS	Special	O	IN/OUT	8	No	2-AC-31	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No, Type B	N/A	Inside	Special Closure	36"	----	----	Closed	----	----	Closed
							No	2-RW-280	Outside	36"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-RW-291	Inside	1/2"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-RW-31	Inside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-RW-292	Inside	1/2"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
43	Reactor Coolant Pump Seals Controlled Bleed Off	CVCS	IA	P	OUT	6	Yes	2-CH-506	Inside	3/4"-Globe-1	3/4"	Diaphragm	CIAS	Open	Closed	Yes	Closed
							Yes	2-CH-198	Outside	3/4"-Globe-1		Diaphragm	CIAS	Open	Closed	Yes	Closed
							Yes	2-CH-505	Outside	3/4"-Globe-1		Diaphragm	CIAS	Closed	Closed	Yes	Closed
							No	2-CH-758, 768, 701	Outside	3/4"-Globe-3		Manual	----	Locked Closed	As Is	No	Closed
							No	2-CH-767, 766	Outside	3/4"-Globe-2		Manual	----	Locked Open	As Is	No	Open
							No	2-CH-744	Outside	3/4"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
47, 69 70, 71	Pressure Monitoring		IA	Special	IN/OUT	30	No	Pene. 47 2-AC-97; Pene. 69 2-AC-99; Pene. 70 2-AC-98; Pene. 71 2-AC-96	Outside	1/2"-Globe-1	1/2"	Manual	----	Open	As Is	Yes	Open
51	Waste Gas Header	RWS	IA	N	OUT	12	Yes	2-GR-11.2	Outside	3"-Globe-1	3"	Diaphragm	CIAS	Open	Closed	Yes	Closed
							Yes[4]	2-GR-11.1	Inside	3"-Globe-1		Diaphragm	CIAS	Open	Closed	Yes	Closed

(99-55)

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification	Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
49	Fire Protection	Fire	IA	O	IN	34	No	2-GR-63	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							Yes	2-Fire-108	Outside	6"-Butterfly-1	6"	Manual	----	Locked Closed	As Is	No	Closed
							No	2-Fire-125	Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							Yes	2-Fire-109	Inside	6"-Check-1		----	----	----	As Is	No	Closed
53	Reactor Bldg. Closed Cooling Water Inlet to Reactor Coolant Pumps and Other Components[8]	RBCCW	IA	N	IN	24	Yes	2-RB-30.1B	Outside	6"-Gate-1	6"	MOV	Remote	Open	As Is	Yes	Open
							No	2-RB-291	Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-RB-173B	Outside	3/4"-Globe-1		Manual	----	Open	As Is	No	Open
54	Reactor Bldg. Closed Cooling Water Outlet from Reactor Coolant Pumps and Other Components[8]	RBCCW	IA	N	OUT	2	Yes	2-RB-37.2B	Outside	6"-Gate-1	6"	MOV	Remote	Open	As Is	Yes	Open
							No	2-RB-300	Outside	1"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-RB-299A	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
85	Containment Leak Rate Pressurization		IA	O	IN/OUT	29	No, Type B	N/A	Inside	6"-Blind Flange-1		----	----	----	----	No	----
							No, Type B	SF-01	Outside	6"-Spectacle Flange-1	6"	----	----	----	----	No	----
							No	2-AC-107	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

(99-55)

TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification		Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position	
61 & 86	Containment Air Sample	CAS	IC	O	OUT	26	Yes	Pene. 61 2-AC-12	Pene. 86 2-AC-47	Outside	1-1/2"-Butterfly-1	1"	Diaphragm	CIAS	Open	Closed	Yes	Closed[10]	(99-55)
							Yes	2-EB-88	2-EB-89	Inside	1-1/2"-Butterfly-1		Diaphragm	CIAS	Open	Closed	Yes	Closed[10]	(99-55)
							No	2-AC-101	2-AC-102	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed	
62 & 87	Containment Air Sample	CAS	IC	O	IN	28	Yes	Pene. 62 2-AC-54	Pene. 87 2-AC-55	Inside	1/2"-Check-1	1"	----	----	----	----	No	----	
							Yes	2-AC-15	2-AC-20	Outside	1-1/2"-Butterfly-1		Diaphragm	CIAS	Open	Closed	Yes	Closed[10]	(99-55)
							No	2-AC-103	2-AC-104	Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed	
67	Refueling Water Purification	RPCS	IA	O	OUT	27A	Yes[4]	2-RW-232		Inside	4"-Gate-1	4"	Manual	----	Locked Closed	As Is	No	Closed	(99-55)
							Yes	2-RW-21		Outside	4"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed	
							No	2-RW-158		Outside	3/4"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed	
68	Refueling Water Purification	RPCS	IA	O	IN	27B	Yes[4]	2-RW-154		Inside	4"-Gate-1	4"	Manual	----	Locked Closed	As Is	No	Closed	(99-55)
							Yes	2-RW-63		Outside	4"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed	
							No	2-RW-159		Outside	3/4"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed	
82	Hydrogen Purge	HC	IA	O	OUT	25A	Yes[4]	2-EB-91		Inside	6"-Butterfly-1	6"	Air Cyl.	CIAS & Hi CTMT Radiation	Closed	As Is	Yes	Closed	(99-55)
							Yes	2-EB-92		Outside	6"-Butterfly-1		Diaphragm	CIAS & Hi CTMT Radiation	Closed	Closed	Yes	Closed	

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List." (99-55)

TABLE 5.2-11

CONTAINMENT STRUCTURE ISOLATION VALVE INFORMATION

Pene. No.	Service	System	Pene. Type[1]	Pene. Category[2]	Flow Direction	Valve Arrgt.	Type C Testing Requirements[3]	Valve Identification		Location Ref. to C.S.	Valve Size-Type-No.	Pene. Line Size	Method of Actuation	Signal	Normal Valve Position	Valve Pos. w/Pwr. Fail.	Pos. Ind.	Post Incid. Position
							Yes[4]	2-EB-86		Inside	3/4"-Gate-1		Manual	----	Locked Closed	As Is	No	Closed
							No	2-EB-120		Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
83	Hydrogen Purge	HC	IA	O	OUT	25B	Yes[4]	2-EB-100		Inside	6"-Butterfly-1	6"	Air Cyl.	CIAS & Hi CTMT Radiation	Closed	As Is	Yes	Closed
							Yes	2-EB-99		Outside	6"-Butterfly-1		Diaphragm	CIAS & Hi CTMT Radiation	Closed	Closed	Yes	Closed
							No	2-EB-121		Outside	3/4"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
65 & 72	Steam Generator Blowdown Sample	SGBS	IA	N	OUT	14	Yes	Pene. 65 2-MS-191A	Pene. 72 2-MS-191B	Outside	1/2"-Globe-1	1/2"	Diaphragm	CIAS	Open	Closed	Yes	Closed
63 & 64	Ctmt. Pressure Test Conn.	ILRT	IC	O	OUT	31	Yes	Pene. 63 2-AC-114	Pene. 64 2-AC-112	Outside	1"-Globe-1	1"	Manual	----	Locked Closed	As Is	No	Closed
							Yes	2-AC-117	2-AC-116	Inside	1"-Globe-1		Manual	----	Locked Closed	As Is	No	Closed
							No	1"-Blind Flange TC	1"-Blind Flange TC	Outside	1"-Blind Flange-2	1"	----	----	----	----	No	----

1 See Figure 5.2-8.
2 See Subsection 5.2.8.2.1.
3 Containment Isolation Valve Test (Type C) per 10 CFR Part 50, Appendix J.
4 Valve tested with pressure applied opposite to that applied during LOCA.
5 Valve 2-CH-435 no longer functions as a check valve; its internal disk and piston spring have been removed.
6 If steam generator pressure drops to ≤572 psia.
7 Valve 2-MS-202 has its closing coil removed, 2-MS-65A/B have their opening coils removed, and 2-SI-651 has its closing and opening coils removed to prevent spurious operation during an Appendix R Fire. See FSAR Section 9.10.6.3, item (2), and Appendix R Compliance Report.
8 "Other components" include Reactor Vessel Support Cooling Coils, CEDM Coolers, and the Quench Tank & PDT HX.
9 The post incident position of 2-IA-566 and 2-IA-27.1 is the same as their normal position unless changed in accordance with plant procedures.
10 Valves 2-EB-88, 2-EB-89, 2-AC-12, 2-AC-15, 2-AC-20, 2-AC-47 may be reopened to permit post-accident hydrogen sampling of the Containment atmosphere. Thus, the sample systems, excluding the Radiation Monitor skids RM8123 & RM8262, are included in the 10 CFR 50, Appendix J testing at the accident pressure.

Note: When making changes to this table, also refer to the Technical Requirements Manual, Table 5-1, "Containment Isolation Valve List."

TABLE 5.2-13

MAJOR[1] CONTAINMENT ISOLATION VALVES

VALVE ID #	PENETRATION NO.	SIZE	VALVE TYPE	BODY MATL[2]	CODE	NUC. CLASS	RATING
2-PMW-43	1	2 in.	Globe	A351, GR CF8M	*	II	150#
2-PMW-3	1	2 in.	Check	A182, GR F-316	*	II	600#
2-CH-006	2	2 in.	Gate	A182, GR F-316	*	II	1500#
2-CH-516	2	3 in.	Globe	A182, GR F-316	*	II	1500#
2-CH-089	2	2 in.	Globe	A182, GR F-316	*	II	1500#
2-CH-515	2	3 in.	Globe	A182, GR F-316	*	II	1500#
2-CH-434	3	2 in.	Gate	A182, GR F-316	*	II	1500#
2-CH-429	3	2 in.	Gate	A182, GR F-316	*	II	1500# (97-189)
2-CH-432	3	2 in.	Check	A182, GR F-316	*	II	1500#
2-CH-433	3	2 in.	Check	A182, GR F-316	*	II	1500#
2-CH-518	3	2 in.	Globe	A351, GR CF-8	*	II	1500#
2-CH-519	3	2 in.	Globe	A351, GR CF-8	*	II	1500#
2-CH-517	3	2 in.	Globe	A351, GR CF-8	*	II	1500#
2-CH-431	3	2 in.	Check	A182, GR F-316	*	II	1500#
2-CH-435	3	2 in.	Check	A182, GR F-316	*	II	1500#
2-CS-5A	4	8 in.	Check	A351, GR CF-8	*	II	300#
2-CS-4.1A	4	8 in.	Gate	A351, GR CF-8	*	II	300#
2-CS-5B	5	8 in.	Check	A351, GR CF-8	*	II	300#
2-CS-4.1B	5	8 in.	Gate	A351, GR CF-8	*	II	300#
2-SI-645	6	6 in.	Globe	A182, GR F-316	*	II	1500#
2-SI-144	6	6 in.	Check	A182, GR F-316	*	I	1500#
2-SI-646	6	2 in.	Globe	A182, GR F-316	*	I	1500#
2-SI-647	6	2 in.	Globe	A182, GR F-316	*	I	1500#

TABLE 5.2-13

MAJOR[1] CONTAINMENT ISOLATION VALVES

VALVE ID #	PENETRATION NO.	SIZE	VALVE TYPE	BODY MATL[2]	CODE	NUC. CLASS	RATING
2-SI-009	6	2 in.	Check	A182, GR F-316	*	I	1500#
2-SI-143	6	2 in.	Check	A182, GR F-316	*	I	1500#
2-SI-706D	6	6 in.	Check	A351, GR CF-8	*	I	1500#
2-SI-615	7	6 in.	Globe	A182, GR F-316	*	II	1500#
2-SI-114	7	6 in.	Check	A182, GR F-316	*	I	1500#
2-SI-616	7	2 in.	Globe	A182, GR F-316	*	I	1500#
2-SI-617	7	2 in.	Globe	A182, GR F-316	*	I	1500#
2-SI-012	7	2 in.	Check	A182, GR F-316	*	I	1500#
2-SI-113	7	2 in.	Check	A182, GR F-316	*	I	1500#
2-SI-706A	7	6 in.	Check	A351, GR CF-8	*	I	1500#
2-SI-636	8	2 in.	Globe	A182, GR F-316	*	I	1500#
2-SI-637	8	2 in.	Globe	A182, GR F-316	*	I	1500#
2-SI-010	8	2 in.	Check	A182, GR F-316	*	I	1500#
2-SI-133	8	2 in.	Check	A182, GR F-316	*	I	1500#
2-SI-706C	8	6 in.	Check	A351, GR CF-8	*	I	1500#
2-SI-635	8	6 in.	Globe	A182, GR F-316	*	II	1500#
2-SI-134	8	6 in.	Check	A182, GR F-316	*	I	1500#
2-SI-625	9	6 in.	Globe	A182, GR F-316	*	II	1500#
2-SI-124	9	6 in.	Check	A182, GR F-316	*	I	1500#
2-SI-626, 2-SI-627	9	2 in.	Globe	A182, GR F-316	*	I	1500#
2-SI-123	9	2 in.	Check	A182, GR F-316	*	I	1500#
2-SI-011	9	2 in.	Check	A182, GR F-316	*	I	1500#
2-SI-706B	9	6 in.	Check	A351, GR CF-8	*	I	1500#

TABLE 5.2-13

MAJOR[1] CONTAINMENT ISOLATION VALVES

VALVE ID #	PENETRATION NO.	SIZE	VALVE TYPE	BODY MATL[2]	CODE	NUC. CLASS	RATING
2-SI-709	10	12 in.	Gate	A351, GR CF-8	*	I	1500#
2-SI-651	10	12 in.	Gate	A182, GR F-316	*	I	1500#
2-SI-463	11	2 in.	Gate	A182, GR F-316	*	II	600#
2-CS-16.1A	12	24 in.	Gate	A351, GR CF-8	*	II	150#
2-CS-16.1B	13	24 in.	Gate	A351, GR CF-8	*	II	150#
2-SSP-16.2	14	3 in.	Globe	A351, GR CF8M	*	II	150#
2-SSP-16.1	14	3 in.	Globe	A351, GR CF8M	*	II	150#
2-FW-5A	15	18 in.	Check	A216, GR WCB	*	II	600#
2-FW-12A	15	6 in.	Check	A216, GR WCB	*	II	600#
2-FW-5B	16	18 in.	Check	A216, GR WCB	*	II	600#
2-FW-12B	16	6 in.	Check	A216, GR WCB	*	II	600#
2-MS-64A	19	34 in.	Check	A216, GR WCB	*	II	600#
2-MS-201	19	4 in.	Gate	A105, GR 2	*	II	600#
2-MS-3A	19	12 in.	Gate	A105, GR 2	*	II	600#
2-MS-190A	19	8 in.	Globe	A216, GR WCB	*	II	600#
2-MS-265B	19	1 in.	Globe	A182, GR F-316	*	II	600#
2-MS-65A	19	3 in.	Globe	A105, GR 2	*	II	600#
2-MS-64B	20	34 in.	Check	A216, GR WCB	*	II	600#
2-MS-202	20	4 in.	Gate	A105, GR 2	*	II	600#
2-MS-3B	20	12 in.	Gate	A105, GR 2	*	II	600#
2-MS-190B	20	8 in.	Globe	A216, GR WCB	*	II	600#
2-MS-266B	20	1 in.	Globe	A182, GR F-316	*	II	600#
2-MS-65B	20	3 in.	Globe	A105, GR 2	*	II	600#

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TABLE 5.2-13

MAJOR[1] CONTAINMENT ISOLATION VALVES

VALVE ID #	PENETRATION NO.	SIZE	VALVE TYPE	BODY MATL[2]	CODE	NUC. CLASS	RATING
2-LRR-61.1	21	1/2 in.	Globe	A351, GR CF8M	*	II	2500#
2-RC-001	21	3/4 in.	Globe	A351, GR CF8M	*	II	2500#
2-RC-002	21	3/4 in.	Globe	A351, GR CF8M	*	II	2500#
2-RC-003	21	3/4 in.	Globe	A351, GR CF8M	*	II	2500#
2-RC-45	21	3/4 in.	Globe	A351, GR CF8M	*	II	2500#
2-MS-220A	22	2 in.	Globe	A216, GR WCB	*	II	600#
2-MS-220B	23	2 in.	Globe	A216, GR WCB	*	II	600#
2-RB-30.1A	24	8 in.	Gate	A350, GR LF1	*	II	150#
2-RB-28.1D	25	10 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.1B	26	10 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.1A	27	10 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.1C	28	10 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-37.2A	29	8 in.	Gate	A350, GR LF1	*	II	150#
2-RB-28.3D	30	10 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.2D	30	6 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.3B	31	10 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.2B	31	6 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.3A	32	10 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.2A	32	6 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.3C	33	10 in.	Butterfly	A516, GR 70	*	II	150#
2-RB-28.2C	33	6 in.	Butterfly	A516, GR 70	*	II	150#
2-SI-312	34	3/4 in.	Globe	A182, GR F-316	*	II	150#
2-LRR-43.1	35	3 in.	Globe	A351, GR CF8M	*	II	150#

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TABLE 5.2-13

MAJOR[1] CONTAINMENT ISOLATION VALVES

VALVE ID #	PENETRATION NO.	SIZE	VALVE TYPE	BODY MATL[2]	CODE	NUC. CLASS	RATING
2-LRR-43.2	35	3 in.	Globe	A351, GR CF8M	*	II	150#
2-IA-569	36	1/2 in.	Check	A182, GR F-316	ASME '83	II	600#
2-IA-566	36	1/2 in.	Gate	A182, GR F-316	ASME '83	II	600#
2-IA-27.1	37	2 in.	Globe	A216, GR WCB	*	II	600#
2-IA-43	37	2 in.	Check	A216, GR WCB	*	II	600#
2-SA-19	38	2 in.	Gate	A105, GR 2	*	II	600#
2-SA-22	38	2 in.	Check	A105, GR 2	*	II	600#
2-AC-4	39	48 in.	Butterfly	A516, GR 70	*	II	150#
2-AC-5	39	48 in.	Butterfly	A516, GR 70	*	II	150#
2-AC-6	40	48 in.	Butterfly	A516, GR 70	*	II	150#
2-AC-7	40	48 in.	Butterfly	A516, GR 70	*	II	150#
2-CH-506	43	3/4 in.	Globe	A351, GR CF8M	*	II	2500#
2-CH-198	43	3/4 in.	Globe	A351, GR CF8M	*	II	2500#
2-CH-505	43	3/4 in.	Globe	A351, GR CF8M	*	II	2500#
2-FIRE-108	49	6 in.	Butterfly	A216, GR WCB	ASME '77	II	150#
2-FIRE-109	49	6 in.	Check	A216, GR WCB	ASME '77	II	150#
2-GR-11.2	51	3 in.	Globe	A216, GR WCB	*	II	150#
2-GR-11.1	51	3 in.	Globe	A216, GR WCB	*	II	150#
2-RB-30.1B	53	6 in.	Gate	A105, GR 2	*	II	150#
2-RB-37.2B	54	6 in.	Gate	A105, GR 2	*	II	150#
2-AC-12	61	1-1/2 in.	Butterfly	A515, GR 70	*	II	300#
2-EB-88	61	1-1/2 in.	Butterfly	A515, GR 70	*	II	300#
2-AC-15	62	1-1/2 in.	Butterfly	A515, GR 70	*	II	300#

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TABLE 5.2-13

MAJOR[1] CONTAINMENT ISOLATION VALVES

VALVE ID #	PENETRATION NO.	SIZE	VALVE TYPE	BODY MATL[2]	CODE	NUC. CLASS	RATING
2-AC-54	62	1/2 in.	Check	A182, GR F-316	*	II	600#
2-MS-191A	65	1/2 in.	Globe	A216, GR WCB	*	II	600#
2-RW-232	67	4 in.	Gate	A182, GR F-316	*	II	150#
2-RW-21	67	4 in.	Gate	A182, GR F-316	*	II	150#
2-RW-154	68	4 in.	Gate	A182, GR F-316	*	II	150#
2-RW-63	68	4 in.	Gate	A182, GR F-316	*	II	150#
2-MS-191B	72	1/2 in.	Globe	A216, GR WCB	*	II	600#
2-EB-91	82	6 in.	Butterfly	A516, GR 70	*	II	150#
2-EB-92	82	6 in.	Butterfly	A516, GR 70	*	II	150#
2-EB-100	83	6 in.	Butterfly	A516, GR 70	*	II	150#
2-EB-99	83	6 in.	Butterfly	A516, GR 70	*	II	150#
2-AC-47	86	1-1/2 in.	Butterfly	A515, GR 70	*	II	150#
2-EB-89	86	1-1/2 in.	Butterfly	A515, GR 70	*	II	150#
2-AC-20	87	1-1/2 in.	Butterfly	A515, GR 70	*	II	150#
2-AC-55	87	1/2 in.	Check	A182, GR F-316	*	II	600#

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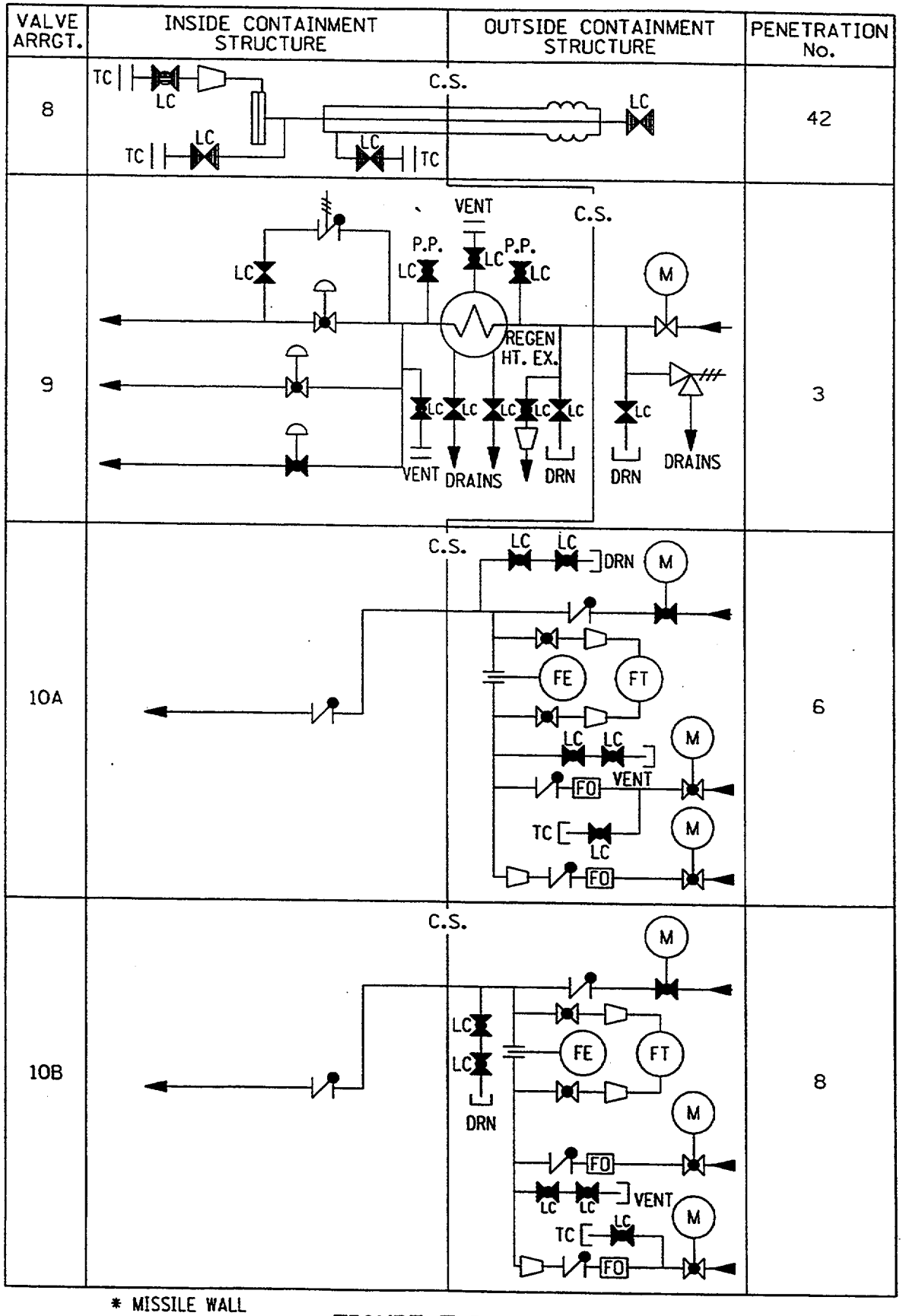
*ASME Section III, 1971, or Draft ASME Pump and Valve Code
Penetration Nos. 41, 46, 48, 50, 52, 55, 56, 57, 58, 59, 60, 66, 73, 74, 78, 79, 80, 81, 84, 88, 89 - SPARE
Penetration No. 85 - Leak Rate Testing
Penetration Nos. 63, 64 - Test Connection

Notes:
1. "Major" valves are principal valves used for containment integrity and process line function (does not include test, vent, drain valves)
2. ASTM material for valve body indicated. Designation is 'A' or 'SA' per controlling code.



* MISSILE WALL

FIGURE 5.2-28
ISOLATION VALVE ARRANGEMENTS

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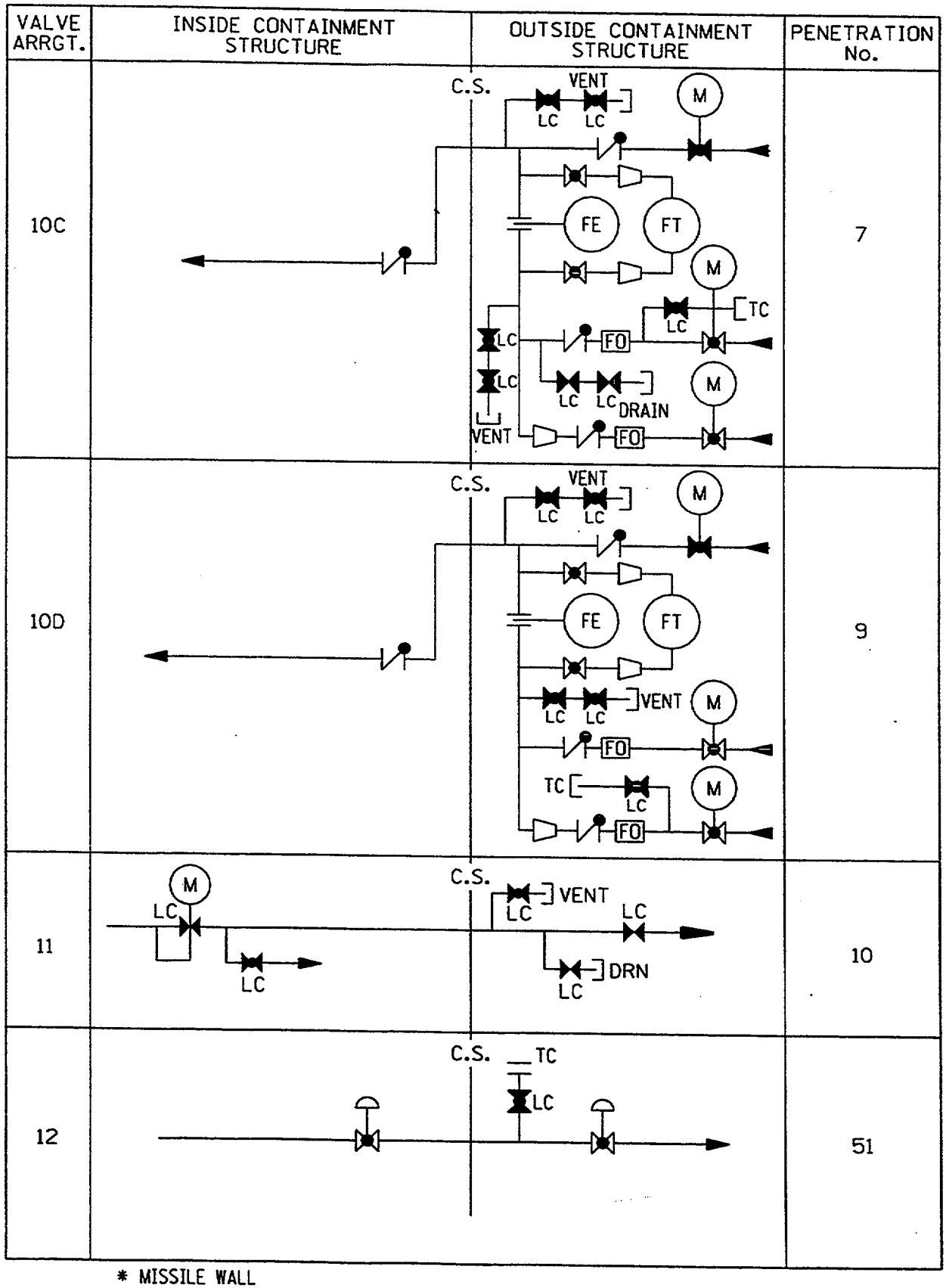

* MISSILE WALL

FIGURE 5.2-29
ISOLATION VALVE ARRANGEMENTS

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f. In order to satisfy the criteria, protection against pipe whip is provided for high energy fluid systems (or portions of systems) except where:

1. either of the following piping system conditions are met (a) the service temperature is less than 200°F, and (b) the design pressure is 275 psig or less, or

2. the piping is physically separated (or isolated) from other piping or components by protective barriers, or restrained from whipping by plant design features, such as concrete encasement, or

3. following a single break, the unrestrained pipe movement of either end of the ruptured pipe in any possible direction about a plastic hinge formed at the nearest pipe whip restraint cannot impact any safety-related structure, system or component, or

4. the internal energy level associated with the whipping pipe can be demonstrated to be insufficient to impair the safety function of any structure, system, or component to an unacceptable level.

5. Pursuant to 10 CFR Appendix A, GDC 4, Leak-Before-Break (LBB) analyses when reviewed and approved by the commission grant exemption from the protection requirements against dynamic effects associated with postulated pipe ruptures. Such an exemption has been granted for the main reactor coolant loop, for the pressurizer surge line, and for the unisolable RCS portions of the safety injection and shutdown cooling piping systems. 99-71

g. Structural failures of piping or components, whether in safety or nonsafety grade systems, in addition to the initial postulated pipe break and its direct consequences, need not be considered. (99-44)

The energy level in a whipping pipe is considered as insufficient to rupture an impacted pipe of equal or greater nominal pipe size and equal or heavier wall thickness.

6.1.4.1.1.2 Postulated Pipe Ruptures

For ASME Section III, Class I piping exceeding the pressure and temperatures conditions indicated in Section 6.1.4.1.1.1 and listed in Table 6.1-1, a pipe rupture is postulated at the following locations:

a. the terminal ends, and

b. any intermediate locations between terminal ends where the primary plus secondary stress intensities S_n (circumferential or longitudinal) derived on an elastically calculated basis under the loadings associated with one-half safe shutdown earthquake and operational plant conditions exceeds 2.0 S_m for ferritic steel, and 2.4 S_m for austenitic steel, and

c. any intermediate locations between terminal ends where the cumulative usage factor (U) derived from the piping fatigue analysis and based on all normal, upset and testing plant conditions exceeds 0.1.

NOTE: The original plant design required a minimum of two intermediate break locations by selecting arbitrary break points when the criteria in Items (b) and (c) failed to provide this minimum number. In adopting Generic Letter 87-11 and eliminating these previously postulated arbitrary intermediate breaks from the population of high energy line breaks postulated for MP2, all existing protective hardware, originally designed to mitigate the effects of these breaks, were retained. Any subsequent removal of this hardware is required to be addressed on a case-by-case basis following the requirements of GL 87-11.

For all other piping systems inside containment exceeding the pressure and temperature conditions indicated in Section 6.1.4.1.1.1 and listed in Table 6.1-1, a pipe rupture is postulated at the following locations:

a. the terminal ends, and

b. any intermediate locations between terminal ends where either the circumferential or longitudinal stress derived on an elastically calculated basis under the loadings associated with seismic events and operational plant conditions exceed 0.8 (S_h + S_A) or the expansion stresses exceed 0.8 S_A. S_A and S_h are defined in Sections NC-3600 and ND-3600 of the ASME Section III Code, Winter 1972 Addenda.

NOTE: The original plant designs required a minimum of two intermediate break locations by selecting arbitrary break points when the criteria in Item (b) failed to provide this minimum number. In adopting Generic Letter 87-11 and eliminating these previously postulated arbitrary intermediate breaks from the population of high energy line breaks postulated for MP2, all existing protective hardware, originally designed to mitigate the effects of these breaks, were retained. Any subsequent removal of this hardware is required to be addressed on a case-by-case basis following the requirements of GL 87-11.

For piping systems outside containment exceeding the pressure and temperature conditions indicated in Table 6.1-4, pipe rupture is discussed in Section 6.1.4.1.1.3.

The type of pipe rupture at any postulated location, whether inside or outside containment, may be either of the following, regardless of the state of stress in the vicinity:

a. Elongated slot type failure in pipes having a nominal diameter of 4 inches and larger. (The equivalent area of which is assumed to be equal to the equivalent internal cross-sectional area of the upstream section of pipe and oriented such that the resultant force is perpendicular to the axis of the pipe.) However, elongated slot type (i.e., longitudinal) breaks are not postulated at terminal ends. Terminal ends are defined as extremities of piping runs that connect to structures, large components (e.g., vessels, pumps, etc.) or pipe anchors that act as essentially rigid constraints to thermal expansion. 99-77

S_p = plastic section modulus

S = elastic section modulus

Equation (1) is presented in the referenced BN-TOP-2 and Equation (2) is used in designing pipe whip restraints for Millstone Unit 2. Since the ratios of the plastic to elastic section moduli range from 1.1 to 1.18 with an average of 1.134 (Reference 6.1-1) for structural shapes, equation (2) becomes:

$$M_u = 0.88\, f_y S_p \qquad (3)$$

Since M_u is the capacity moment for the restraint and is associated with the energy absorbing capacity of the restraint, the lesser values of M_u used in Millstone Unit 2 will result in a more conservative design than that if equation (1) is used.

(3) Reference: Pipe break configurations are shown on Figure 2-4 of BN-TOP-2. The referenced BN-TOP-2 figure and the associated discussion on jet impingement configurations and forces attribute jet impingement effects not only to full size circumferential or longitudinal breaks, but also to cracks. Millstone Unit 2 does not consider any jet impingement effects or other dynamic consequences resulting from pipe cracks. Through-wall leakage cracks in high energy piping are evaluated for environmental consequences only (i.e., temperature, humidity, and flooding effects).

99-77

As noted within the BN-TOP-2 report, jet impingement effects are important only in the first few break diameters away from the source pipe. Since the effective diameter of a crack is based on a circle with an area equal to one-half the pipe's inside diameter times one-half the pipe's wall thickness instead of the full pipe flow area considered for other breaks, its effective diameter is significantly smaller than that of other breaks. As both the effective zone-of-influence and resulting jet force due to any break are a function of the break diameter, the dynamic effects resulting from a crack would also be significantly less than full size break effects. Therefore for Millstone Unit 2, jet impingement effects resulting from cracks are considered fully enveloped by the postulated full size pipe breaks.

6.1.4.1.1.3 Pipe Ruptures Outside Containment

Rupture Criteria

For all piping systems with normal operating pressure and temperature conditions exceeding those defined by operating conditions A, B and C in table 6.1-4, a pipe rupture is postulated at the following locations:

1. the terminal ends, and

2. for stress analyzed piping, at any intermediate location between terminal ends where the calculated Primary + Secondary Stress, S, exceeds $0.8(S_h + S_A)$, or the Secondary Stress alone exceeds $0.8S_A$ where:

   S = stresses under the combination of loadings associated with the normal and upset plant condition loadings and an OBE event.

S_h = basic material allowable stress at maximum (hot) temperature from the allowable stress table in the associated piping analysis design Code.

S_A = allowable stress range for expansion stresses as defined in the associated piping analysis design Code.

3. for unanalyzed piping, at all intermediate weld connections to elbows and other pipe fittings within the line.

4. for type and configuration of the resulting postulated pipe ruptures at each of the above locations, refer to Section 6.1.4.1.1.2,

5. In addition to the above pipe ruptures, through-wall leakage cracks are postulated at those locations that results in the most adverse conditions of flooding and local environmental conditions on the adjacent safe shutdown equipment. The flow area of leakage cracks is taken as the area equal to that of a rectangle one-half the pipe inside diameter in length and one-half the pipe wall thickness in width.

Plant Modifications Resulting From the Elimination of Arbitrary Intermediate Breaks (GL 87-11)

In adopting Generic Letter 87-11 and eliminating the previously postulated arbitrary intermediate breaks from the population of high energy line breaks postulated for MP2, all existing protective hardware, originally designed to mitigate the effects of these breaks, were retained. Any subsequent removal of this hardware is required to be addressed on a case-by-case basis following the requirements of GL 87-11.

6.1.4.1.1.4 Safe Shutdown for Pipe Rupture (98-3)

A comprehensive review of MP2 design basis documentation has been performed and a methodology was developed to select safe shutdown paths. Those components necessary to bring the plant to Mode 3, Hot Standby, condition are defined by this methodology. The various alternative methods identify paths used in accomplishing the following safe shutdown functions subsequent to a High Energy Line Break (HELB):

- RCS Heat Removal
- RCS Pressure Control
- RCS Inventory Control
- Maintaining Vital Auxiliaries

General assumptions used in identifying the safe shutdown paths are described below:

- Safe shutdown for a HELB is defined as Hot Standby (Mode 3) and maintaining this condition for eight (8) hours.
- Only those high energy systems that are in service during normal plant operations are assumed to rupture.
- Offsite power is assumed to be not available if the pipe rupture results in a unit trip.
- Coincident with the postulated pipe break, a single active failure in an active component of a required safety related system is assumed.

- No other extraordinary events are postulated (i.e., fire, earthquake, etc.) (98-3)

6.1.4.1.2 Design Method for Damage Prevention

Physical separation, where possible, between critical lines is utilized by routing the piping so that a ruptured line is beyond reach of another line or critical component.

The safety injection and containment spray pumps arranged in two groups of minimum safety features, each group consisting of one high pressure safety injection pump, one low pressure safety injection pump and one containment spray pump.

Each group is served by separate suction and discharge header, powered by separate sources and is physically located in separate watertight pump rooms. The third high pressure safety injection pump is interconnected to the two headers. A selection switch is provided to enable the pump to draw power from either emergency bus. The third HPSI pump is aligned to operate only upon removal from service of either of the other HPSI pumps.

Water hammer effects are minimized in the engineered safety features system by keeping process lines flooded where possible. Air pockets will be eliminated by periodic opening of isolation valves during testing and by high point vents provided on the process lines.

The safety injection system is flooded up to the connection to the reactor coolant system while the containment spray system is flooded up to the risers in containment. The remaining piping is vertical or sloped upward to eliminate air pockets.

All motor-operated valves in the safety injection and containment spray systems are opened in response to their associated actuation signals.

The 8 HPSI injection throttle valves are manually positioned to the desired open position and do not open further on an actuation signal.

Further discussion of the various methods of pipe whip protection are contained in Subsection 5.2.6.7.1.

The following criteria must be satisfied by components located in the containment and the enclosure building that are not seismic category I: (97-98)

a. Failure of these components cannot affect safe shutdown of the reactor.

b. Failure of these components cannot affect the operation of Seismic Category I components.

c. Nonseismic category I components are located in such areas that missiles projected from their failures remain isolated from seismic I components.

Table 6.1-3 lists the nonseismic category I components located inside the containment and enclosure building.

The components of the clean liquid waste processing systems are located at the lower elevations of containment, away from safety related equipment. The pressurizer quench tank operates at 35 psig maximum. The primary drain tank operates at 2 psig. At these relatively low operating pressures, these components do not sustain sufficient energy to become potential missiles. The cooler which is connected to the RBCCW is analyzed for the design basis earthquake to assure integrity.

The vents, drains and sample connections to the reactor coolant pressure boundary are analyzed for the design basis earthquake beyond the second isolation to the first anchor.

The containment floor drain system is embedded in concrete and vented to containment. These are atmospheric pressure and therefore, do not sustain sufficient energy to become a potential missile or to whip into category I equipment. The sump pumps are located at the lowest elevation of containment, away from safety related equipment. The discharge pressure of the sump pumps is approximately 25 psig. This pressure is not sufficient to create a potential missile or to whip.

The components of the RV Head decontamination system are located at lower elevations in shielded compartments away from safety related systems. The system is no longer used and therefore will not generate any missiles due to rupture. (97-147)

The containment elevator is completely contained within the elevator shaft structure which is a Category I structure. The stud storage racks are anchored during normal operation. The CEDM coolers are connected to the RBCCW and are analyzed for the design basis earthquake. The CEDM coolers are located on the missile shield, away from any safety related equipment. The revolving maintenance truss is analyzed for the design basis earthquake. The containment auxiliary circulation fans and duct work are located above the operating floor, away from safety related equipment. (97-48)

The instrument air receiver tank, which operates at a pressure of 80 psig, is located at elevation (+) 14'-6" outside of the secondary shield wall.

The refueling pool skimmer filters are located in shielded compartments at elevation (+) 14'-6" outside the secondary shield wall. The skimmer system is in operation only during refueling.

The reactor coolant system is shielded from potential missiles by the primary and secondary shield walls and the missile shield. The steam generators are shielded by the secondary shield walls and by a doghouse extended above the operating floor.

The blowdown quench tank, heat exchanger and pump are located within a concrete enclosure through which the main steam piping is routed. The blowdown tank is located within the enclosure building proper and is normally operating at approximately 1 psig. At this low pressure the system does not sustain sufficient energy to become potential missiles or to cause the pipe to whip. The closed quench tank system operates at approximately 50 psig. This energy is not sufficient to cause damage to the Category I systems following a postulated pipe whip.

Miscellaneous duct work and unit heaters which operate at relatively low pressure shall not attain sufficient energy to cause damage to category I systems.

6.1.4.2 Tests and Inspection

The engineered safety features system components are procured to detailed engineering specifications and tested to applicable codes as described in the subsequent section. In addition, special testing of actual or prototype components under proposed service conditions of pressure, temperature and humidity are conducted on various components. Engineered safety features system components are incorporated with provisions for on-line testing.

Engineered safety feature systems are incorporated with provisions for full system actuation and performance tests.

Components located outside the containment are accessible for periodic maintenance and inspection. Components located inside the containment are accessible during normal shutdown.

REFERENCES

6.1-1 Plastic Design in Steel, 2nd Edition, ASCE Manual No. 41, 1971. (97-92)

6.1-2 NUREG/CR-2913, "Two-Phase Jet Loads," January 1983. (98-3) (98-9)

recirculating piping inside the containment extends approximately 11 inches above the floor Elevation (-)22-6 to prevent any clogging due to the accumulation of debris, etc., near or into the pipe. A vortex breaker is provided on each containment sump recirculation pipe inlet to avoid any air locks.

The containment sumps are located at the bottom of the containment building at elevation (-)22-6 and are enclosures around the recirculation pump suction pipes. The sump screen enclosure is a screen assembly constructed of a stainless steel frame and galvanized grating to which a stainless steel wire cloth is attached. The stainless steel wire mesh has 0.080 inch diameter wire with 3/32 inch openings preventing larger particles to pass and thus avoiding any clogging of the containment spray nozzles or HPSI throttle valves. The top of the sump enclosure is a solid stainless steel plate and each sump area is provided with a manway access and vent.

The screen assembly is divided into two sections separated by mesh and grating. A failure of one side would not negate the operation of the remaining assembly.

Due to the low-face velocity across the sump screen, loose objects generated by a pipe break will either settle on the sump screen prior to sump operation or not be effected by sump operation. Loose objects most likely to block the sump screen would be sections of pipe insulation dislodged by a loss-of-coolant accident (LOCA). All insulation debris generated from a LOCA in the steam generator cubicle (worse case) is assumed to be fine fragments. For the design basis with one LPSI pump failing to trip on initiation of recirculation switchover, insulation can be transported to and adhere on the sump screen. However, if no LPSI pump is operated in the recirculation mode, no insulation is expected to build up on the screens because of the low transport velocity.

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The RWST capacity is adequate to provide a minimum water level in the containment sump for operation of the safety injection and containment spray pumps during post-accident operation for both LBLOCA and SBLOCA.

Engineered safety features piping connected to the containment sump is an extension of reactor containment during the recirculation mode of core and containment cooling. The following items pertain to suction piping from the sump to the first isolation valve:

a. Both spool pieces are designed, fabricated, tested and inspected in accordance with ANSI B31.7, Class II. The valves are manufactured with the ASME Code for Pumps and Valves for Nuclear Power, Class II.

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7.3.1.2.7 Testability

The ESAS system is designed to permit testing up to and including the actuation module during power operation.

The ESAS system has an automatic testing system. This system is described in Section 7.3.4.

7.3.1.2.8 Response Time

The overall response time of the ESAS system shall be measured from the sensor input to the trip device output terminals and will not exceed 500 milliseconds.

7.3.2 System Description

7.3.2.1 General

The ESFAS detects accident conditions and initiates the safety features systems which are designed to localize, control, mitigate, and terminate such incidents. The ESAS was designed and constructed by Consolidated Controls Corp. in Bethel, Connecticut. The AFAIS equipment was supplied by the Foxboro Company. The engineered safety features actuation system is divided into four sensor channels (A, B, C, D), two actuation channels and actuation logic channels.

Each of the two ESAS actuation and logic channels includes an automatic load sequencer for sequentially loading the emergency diesel generators (DG) following a loss of normal power. A separate third channel is incorporated into the ESAS to actuate equipment that can be energized by either of the two electrical divisions. 99-88

All process variables are transmitted as analog signals. Loss of voltage on the 4.16-kV emergency bus is detected through the potential transformers by the ESFAS undervoltage modules.

Four essential or vital power sources are provided for the ESFAS. Two emergency DGs are provided to supply power to the actuated equipment of the protective systems in case of loss of offsite power.

If offsite power is available, the engineered safety features (ESF) equipment starts directly. If offsite power is not available, load shedding and sequencing are required for sequential loading of the DGs.

As a result of IE Bulletin No. 80-06, "Engineered Safety Feature Reset Controls," the ESF Reset Controls have been modified so that ESF actuated equipment remains in its emergency, "actuated," mode following reset of an ESFAS until deliberate operator action is taken.

7.3.2.2 Sensor Channels

The instrument channels monitor redundant and independent process variables and initiate a sensor channel trip when the variable or condition deviates beyond a set limit. Each of

following actuation channels are initiated by the ESFASs: (See Figure 7.3-1, 7.3-3 and 7.3-4.)

a. SIAS under the condition: low-low pressurizer pressure 2/4 (or 2/3) or high containment pressure 2/4 (or 2/3) or manual SIAS (main control board push-button). |99-88

b. CSAS under the following conditions: SIAS (manual or automatic) and high-high containment pressure 2/4 (or 2/3) or manual CSAS (main control board push-button).

c. CIAS for automatic containment isolation valves under the conditions: low-low pressurizer pressure 2/4 (or 2/3) or high containment pressure 2/4 (or 2/3) or manual SIAS or CIAS (main control board pushbuttons).

d. EBFAS under the following conditions: low-low pressurizer pressure 2/4 (or 2/3) or high containment pressure 2/4 (or 2/3) or manual EBFAS or SIAS (main control board pushbuttons). |99-88

e. Containment purge valves close signal (CPVIS) under the following conditions: high containment radiation 1/4.

f. Containment hydrogen purge valves close signal under the following conditions: CIAS. In addition, two high range gamma-sensitive ion chambers are used to detect post-accident radiation levels in the containment. The ESAS CIAS logic in conjunction with the logic from the post-accident radiation monitors is combined together in an "OR" logic to initiate closure of the containment hydrogen purge valves. The containment hydrogen purge valve closure is generated either by one-out-of-two post-accident radiation monitors exceeding its setpoint, or by an ESAS CIAS signal.

g. MSIAS upon low steam generator pressure 2/4 (or 2/3) or high containment pressure 2/4 (or 2/3) or manual MSIAS (main control board push buttons).

h. SRAS for the following conditions: low-low RWST level 2/4 (or 2/3) or manual SRAS (main control board pushbutton).

i. AEAS for the following conditions: high radiation in the fuel handling area 2/4 (or 2/3) or manual AEAS (main control board pushbutton or local pushbutton) in the absence of EBFAS.

j. Emergency bus load shed for the following conditions: Level 1 emergency bus undervoltage (loss of voltage), 2/4 (or 2/3) channels. FSAR Section 8.2 discusses the automatic load shedding of the emergency buses.

k. Emergency DG start for the following conditions: Level 1 emergency bus undervoltage (loss of voltage), 2/4 (or 2/3) channels or on a SIAS. FSAR Section 8.3 discusses the automatic start of the emergency DGs. |99-88

l. Trip of the RSST supply breaker to the associated 4160 Volt emergency bus for the following conditions: Level 2 emergency bus undervoltage (degraded voltage), 2/4 (or 2/3 channels).

7.3.2.8 Automatic Sequencer

If offsite power is available, the equipment actuated by ESFAS signals is started as soon as the actuation signal is developed by the coincidence logic. When offsite power is not available, the equipment is started by a sequencer. Emergency DGs are provided for supplying power to ESFs in case of loss of offsite power. The load shedding and sequential actuation system initially block, and then unblocks in programmed steps, the actuation channels for the equipment requiring power to perform its intended function. The undervoltage system contains four redundant sensors. The trip outputs are delayed and, by means of two out of four coincidence logic, load shed and diesel start are initiated.

The diesel load sequencers (1/channel) provide 10 independent (5/sequencer), time-separated loading signals as detailed in Figures 7.3-2A through 7.3-2D, and Tables 8.3-2 and 8.3-3.

The sequencer output signals provide the required time sequencing of the safety features and the loss-of-power shutdown equipment according to Figures 7.3-2A through 7.3-2D, and Tables 8.3-2 and 8.3-3.

Means for testing of the sequencers are provided.

Indicators located in the appropriate cabinet will show the state of each sequencer time step.

7.3.2.9 Annunciation

The ESFAS provides contact outputs to indicate status in the control room. The outputs are isolated so that faults occurring external to the actuation system cabinets will in no way degrade the protective function of the system.

The contacts provided are summarized, as follows:

a. Bistables

A contact for each of the 52 ESAS sensor channel trip bistables, to indicate bistable trip, for use on individual windows of the main control board annunciation system.

A contact for each of the four ESAS sensor channel parameters to indicate bistable bypass and conversion from 2/4 to 2/3 logic. | 99-88

A contact from each of the redundant AFAIS actuation logic trains to annunciate a channel in bypass and that an actuation train is converted from a 2/4 to a 2/3 logic.

AFAIS bistable status is indicated by lamps on the status panel for each actuation channel on C517 and C518. Each panel has 4 status lamps, one for each of the four bistable inputs to the actuation channel. The panel for each actuation channel also indicates the "failed/not failed" status of the actuation channel microprocessor. | 99-88

b. ESFAS Actuation

A contact for each of the two channels for the following actuation signals to indicate an actuation trip. The actuation signals alarmed are SIAS, CIAS, CSAS, EBFAS, MSIAS, SRAS, AEAS and AFAIS. A contact for each of the actuation channels for the pressurizer low low pressure, mainsteam isolation to and AFAIS to indicate manual blocking of the channel. 99-88

A contact for each of the actuation channels for both the pressurizer low-low pressure and main steam isolation to indicate permission to block the actuation channel.

c. Sequencers

One contact per sequencer for indication of the automatic sequencing in progress.

One contact per sequencer for indication of sequencer malfunction.

7.3.2.10 Cabinets

a. The ESAS cabinets are located in the main control room behind the main control board (see Figure 7.6-1).

b. The AFAIS circuitry is located in cabinets C517, C518 and SPEC 200 Racks, RC-30A, B, C, D, A-1, B-1 in the main control room behind the main control board (see Figure 7.6-1). 99-88

c. The ESAS cabinets have hinged rear doors and open fronts with semiflush mounted devices.

d. Cabinets RC-30A, B, C, D, A-1, B-1 have hinged front and rear doors, removable side panels with nest mounted modules. C-517 and 518 contain channel status lamps and key operated channel bypass switches.

e. Any adjustment required for routine calibration is accessible from the cabinet front. Adjustments for routine calibration do not require removal of equipment from its housing. Adjustment controls have keyless locking mechanisms and/or they are recessed in the cabinet or applicable chassis.

f. Nameplates are used to identify all cabinets, channels, and assemblies. Individual channels are identified by color code throughout the system.

7.3.2.11 Analog Bistables

The ESAS analog comparator bistables are designed to meet the following specifications:

a. The trip-point adjustment of the analog channel bistables covers the input signal range from 0 through 100 percent. The adjustment is by means of a calibrated dial. The resolution of adjustment is ±0.25 percent when suitable external measuring equipment is used in connection with setting the trip point.

a.	Pressurizer pressure	
	Range (psia)	1500-2500
	Input (volts, DC)	1-5
b.	Containment pressure	
	Range (psig)	60-0
	Input (ma, DC)	4-20
c.	Steam generator pressure	
	Range (psia)	0-1000
	Input (volts, DC)	1-5
d.	Fuel handling area radiation	
	Range	.1 mr/hr to 10^4 mr/hr logarithmic
	Input (volts, DC)	5-1
e.	Refueling water storage tank level	
	Range (%)	0-100
	Input (ma, DC)	10-50
f.	Emergency bus voltage	
	Range (vac)	0-120
	Input (vac)	0-120
g.	Steam generator water level	
	Calibrated range (inches of water column)	173.5-41.4
	Input (ma, DC)	4-20
h.	Containment radiation (Gaseous and particulate)	
	Range	10 cpm to 10^6 cpm
	Input (volts, DC)	5-1

Isolated DC outputs from each sensor of Items b and e above are provided for main control board indications.

Pressure transmitters for items a, b and c above are shared for the RPS and the ESFAS.

The system has output relays that provide electrically isolated contacts for actuation of equipment as tabulated on Figures 7.3-2A through 7.3-2D.

solid conditions. The 340 psia was chosen since this is the maximum expected RCS pressure when the Shutdown Cooling (SDC) System relief valves are aligned to the RCS. Since the calculated peak RCS pressure was less than the Appendix G allowable pressure, no credit was taken for the PORVs to mitigate the energy addition transient.

b. Mass Addition - This transient results from the inadvertent start of a HPSI and/or a charging pump during RCS water solid conditions. The maximum flow from the allowed pump combinations was assumed to occur concurrent with the loss of decay heat removal capabilities and the energizing of the pressurizer heaters. These conservative assumptions in combination with the pump restrictions discussed below, resulted in system peak pressures which are lower than the allowable 10CFR50 Appendix G limits:

1) A maximum of one HPSI and two charging pumps may be capable of injecting into the RCS when the RCS cold leg temperature (Tc) is 190°F < Tc ≤ 275°F or when Tc is < 190°F when the RCS is vented through a 2.2 in^2 passive vent.

2) When the RCS is not vented with a vent size of ≥ 2.2 in^2, a maximum of one charging pump may be capable of injecting into the RCS when Tc ≤ 190°F.

7.4.8.3 System Operation

During plant cooldown, the Appendix G pressure/temperature limit curves are used to decrease the RCS pressure and temperature down to 300°F and 400 psig. Prior to cooling the RCS below 275°F, normal operating procedures require the activation of the PORV "Low setpoint at ≤400 psig by resetting the hand switches to the "Low" position. To assure that the PORVs are reset to the "Low" set point prior to cooling the RCS to ≤275°F, an alarm (reset to low) is activated when the RCS temperature approaches 280°F. Upon resetting a PORV handswitch to "low" (i.e., LTOP mode), the respective motor operated block valve (2-RC-403 or 2-RC-405) upstream of the PORV receives an open signal. This ensures that the PORVs have not been isolated and are capable of performing their function. While the PORV "Low" setpoint is at ≤400 psig, the overpressure transient alarm is activated when the RCS temperature is below a preselected value of ≥275°F and the RCS pressure exceeds a preselected value of ≤360 psig. The purpose of this overpressure transient alarm is to alert the operator that pressure is increasing, and action to control pressure should be taken to preclude PORV actuation. 99-56 99-56

During plant heatup, normal operating procedures require the RCS pressure be maintained below 400 psig until the RCS temperature is greater than 275°F to preclude coolant discharge through the PORVs. When the RCS temperature exceeds 275°F, normal operating procedures require the operator to reset the PORVs to the "High" setpoint relief of approximately 2385 psig. The low temperature transient alarm is also de-energized at this time. After the PORVs are reset to the "High" setpoint of approximately 2385 psig, normal plant heatup continues accordingly.

The start of an RCP significantly changes the flow characteristics in the RCS from one of stagnant or low flow conditions to that of forced/high flow conditions. When no

RCPs are in operation and SDC is in service, the vessel inlet temperature is best represented by the temperature reading of the SDC return line water temperature since little mixing may occur between the hotter RCS coolant and the colder SDC coolant. However, once the first RCP is started, the vessel inlet temperature is best represented by the RCS cold leg temperature readings since the SDC flow mixes with the remainder of the reactor coolant. (97-127)

The start of the first RCP may result in a step temperature increase which may be further amplified by the increase in RCS energy resulting from the added RCP heat input. To minimize the temperature differential between the SDC return coolant temperature and the RCS cold leg coolant temperature, it may be desirable to reduce the SDC heat removal rate which may result in a small RCS temperature increase. Since the accumulation of residual heat in the RCS may also result in increased RCS pressure, the energy addition evaluations assumed that no decay heat would be removed from the RCS for a period of up to 5 minutes following the start of an RCP. This bounding assumption will allow the operators to re-establish the required SDC flow following the start of an RCP without exceeding the allowable pressure/temperature limits.

7.5.6.3.2 System Description

7.5.6.3.2.1 System

The location and types of radiation monitors provided are listed in Tables 7.5-6 and 7.5-6A and described below.

7.5.6.3.2.1.1 Unit 2-Stack Gaseous and Particulate Monitoring

A representative sample is continuously extracted from the Unit 2 stack by one of three isokinetic nozzles. Selection of sample nozzle is automatic, depending on the stack flow rate, as determined by the number of fans in operation.

The air extracted from the stack is directed through an off-line particulate and gaseous monitoring systems with particulate and iodine grab samples for laboratory analysis.

A beta scintillation detector placed in the gaseous sample chamber detects and measures the radioactivity present in the sample volume passing through the chamber. A beta scintillation detector measures the radioactivity of the particulate filter. 99-81

A second monitoring assembly located at elevation 31 feet, six inches in the switchgear room in the turbine building is designed to measure high range post-accident gaseous releases. Additionally, this equipment will sample for particulates and iodine. A required sample flow is used to minimize particulate and iodine buildup. This flow is monitored in a gas sample chamber using two geiger-mueller detectors. Three separate particulate/iodine assemblies are monitored by a geiger-mueller detector for personnel protection and to ensure capability of laboratory counting. The cartridge assemblies sample in a sequential manner such that when one collects the maximum amount of radioactivity specified, the flow path is then switched to the second cartridge assembly. The cartridge assemblies are removed for laboratory analysis.

The high range effluent monitor has both local and remote (control room) indication and control. A recorder in the control room continuously records high range effluent radiation levels.

The high range effluent monitor is electrically interconnected to the normal range vent-stack monitor such that a high range release will put the normal range monitor into a purge mode for the duration of the high range release.

7.5.6.3.2.1.2 Containment Gaseous and Particulate Monitoring

Two redundant off-line Seismic Class I particulate and gaseous monitoring and halogen sampling systems are used to monitor the containment atmosphere by extracting representative samples continuously from the containment atmosphere. The sample first passes through a fixed particulate filter, then passes through an iodine filter cartridge and then into a gas chamber before being exhausted back into the containment atmosphere.

The particulate and the gaseous channels are monitored continuously. The iodine cartridge is removable for periodic laboratory analysis.

- trunked radio system | 98-174
- on-site hand-held and mobile UHF radio repeater security system;
- Connecticut Valley Electric Exchange (CONVEX) Command Control Network (CCN); | (3/91)
- Waterford Police System;
- Tri-town UHF radio system;
- State Police system;
- Very High Frequency (VHF) radio paging system. | (98-155) (3/91)

A dedicated radio remote control console is provided in the Millstone 2 control room for communications with all associated onsite as well as offsite radio facilities (as outlined above). Its power source is lighting panel LPCI, a highly reliable non-vital panel powered by the Emergency Diesel Generator backed Computer Power Inverter D50A. Normally, all radio systems, except the unit's O&M system, are quiet to the unit operator unless selected by the operator for monitoring or operation. Tone alert, except on the O&M system, is provided to enable remotely located radio dispatchers to contact the control room operator. | 98-174

The radio console installed in the Millstone 2 control room consists of two individual bays secured together as a consolidated unit. The total length of the equipment is 46 inches with a height of 43-3/4 inches and an overall depth of 29-1/2 inches. The console is an equipment enclosure housing audio amplifiers (T/R modules), tone generators (encoders), tone decoders, and dual power supplies.

The console generates low level audio and DC voltages only, for the single purpose of controlling remotely located base station radios.

The radio control center equipment is mounted in a single width housing with a beveled front, projected writing surface and panel turret.

The power supplies and termination panels for the control consoles are located in the lower portion of the equipment housing. Provisions are included on the rear-hinged termination panel for securing cable entries. The two upper bays contain the heart of the console radio control system. These two bays can be considered as left center and right center as viewed from the operator side of the console. The left side of the console contains the controls for police, site security, tri-town, operations/maintenance radios, and master control module. The emergency alert paging system occupies the right half of the console. The radio control panel is mounted directly in front of the radio dispatcher's position. This provides the interface functions required between the operator and the console (microphone, speakers, volume controls, push to talk switches).

The microphone is a moving coil, dynamic unidirectional, that is uniform with frequencies of 80 to 13,000 Hz. The microphone is adjustable vertically and horizontally to accommodate different operators and is internally rubber-vibration-isolated to avoid physical damage.

The console contains two power supplies - a low voltage supply and a high voltage supply, each with an input voltage of 120 V AC. The low voltage supply provides +24 V DC and is capable of handling up to 24 radio channels. It includes a nominal +13.8 V DC ±10 percent regulator which, in conjunction with an overcurrent protection circuit, can provide a maximum continuous output current of 1 amp. The power supply has an output current capability of 8 amps. The high voltage supply provides +175 V DC for keying up to 24 DC controlled radios.

The console contains 15 audio amplifiers (T/R modules) with expansion capability of 15 future modules. One T/R module is used with each radio control channel. The module contains both logic circuits and receive/transmit audio circuits. The logic circuits include channel select, keying, busy, and priority functions. The receive audio circuits include speech processing (using an audio compressor circuit), muting, audio gating, and a voice enabled call indicator. The transmit audio circuits include a preamplifier, tone mixing amplifier, gating circuits, and a transmit audio line driver.

There are three tone generators (encoders) with external pushbutton operator controls located in the console. A touch tone encoder allows standard touch code 2-frequency tone codes to be transmitted from the communications console. It can be used wherever a coded signal is required for selective calling or data transmissions. The encoder front panel includes a light emitting diode (LED) indicator which alerts the operator that the transmission of a code can proceed. A programmable timing circuit automatically resets the encoder and unkeys the transmitter if the tone sequence is not entered within a predetermined time. The encoder and transmitter automatically reset if the operator fails to complete a code entry. All codes generated by the encoder are compatible with standard touch tone equipment. A two-tone sequential tone generator allows encoding pocket pagers and fixed receive monitors. The operating controls and indicators are located on the front panel of the unit. The encoder has 16 pushbuttons, a four-digit call code display, a call indicator, and a talk indicator. A code is manually punched into the encoder keyboard and the sequence is automatically sent whenever desired. The transmitter stays on the air for a predetermined amount of time after the code transmittal terminates in order for a voice message to be sent out to the desired pager or monitor. (3/91)

The remaining tone generator is the auto page encoder module. This unit is an engineered hybrid using digitally synthesized tones and logic switching circuits. The encoder, once activated, automatically selects predetermined transmitters and send out a programmed tone sequence to alert pagers and

fixed receive monitors. The unit works in unison with a tape deck containing a canned tape message selected by the operator.

The console contains eight touch tone decoders that activate indicator lights and a sonalert audible device. This alerts the operator of an incoming call on a particular channel. The audio circuits of the console are muted until activated by the decoder, and turned on to normal volume when a proper code sequence is decoded. The audio circuit and sonalert have to be manually reset ensuring that the operator does not miss an incoming call.

It should be noted that the Millstone 2 console is operationally identical to those in Units 1 and 3. This provides an expanded backup system for the communications system on the Millstone site.

O&M Radio

This system is controlled by the consoles in Units 1, 2, and 3. The system consists of a control/base station and a repeater relay station. Control/base station description and location is the same as CCN. The cable description and location is also the same as CCN. The antenna is a fiberglass enclosed, stacked collinear array omnidirectional pattern, capable of a 5 dB gain. The antenna is direct ground lightning protected and has a wind survival of 100 mph. The antenna is capable of an 8 MHz bandwidth to satisfy the frequency requirement of the control/base station. The installed antenna weighs 4 pounds and is 80 inches in length.

The repeater relay station is installed in the Millstone Unit 2 Maintenance Shop (Building 211). The station is fully solid-state, incorporating integrated circuits located on modular plug-in circuit boards. The station is protected for overcurrent conditions and power surges. The station is r-f activated and contains thermal protection on the power amplifier. The primary power source is 120 V AC.

Unheated temperature compensated plug-in oscillator modules are used for frequency control. The station has a continuous duty transmitter that can operate on full power. The station is connected to the antenna via a 7/8-inch jacketed semirigid coaxial cable. The antenna is a 14 element 13 dB gain dual YAGI with a forward controlled pattern. The antenna weighs 8 pounds and has a wind survival of 100 mph. The antenna is direct ground lightning protected.

Trunked Radio System

This is a five channel digital based trunking radio system. The trunking repeaters and central controllers are installed in the Telecommunications Radio Housing adjacent to the Millstone stack. The repeater stations are capable of 75 watts of RF power output with continuous duty operation. The primary power source is 120 V AC. The backup power is provided by a diesel generator. Panel type antennas are mounted on the stack. A 98-174

backup system, designed to come on line in the event of a failure at the primary site, is installed in the Technical Support Building. The Control Room has access to a dedicated channel from the communications console. The base station is located in the CPF Building. The directional antennas are located on the roof of the CPF Building and the installation design is such to withstand windspeeds up to and including 100 mph. 98-174

Security Radio System

This radio system consists of two repeater/relay station and six control/base stations. System control is from the radio consoles in Units 1, 2, and 3, the security central and secondary alarm stations, and the Emergency Operations Facility (EOF). 98-174

The repeater station is capable of 75 watts of power with continuous duty operation. The station is protected against lightning, overcurrent conditions, and power surges. The primary power source is 120 V AC and draws little current. 98-174

The security control/base station is a compact two-way radio suitable for desk top mounting. It fully utilizes the advantages of solid-state circuits; reliability, small size, ruggedness, and low maintenance requirements. The station's primary power source is 120 V AC, and is protected from overcurrent conditions. The stations are connected to the antennas, via a 1/2-inch jacketed semirigid coaxial cable. The antennas are omnidirectional. The antennas have a wind survival of 150 mph and are direct ground lightning protected. 98-174 98-174

Multi-Frequency UHF Repeater System

A UHF repeater system is dedicated to plant operations and maintenance activities. In the event of repeater failure, a "talk around" feature can be accessed from the control room radio console and allows limited communications to continue without the repeater. The radio console is able to access similar but separate UHF radio repeater systems at Millstone Units 1 and 3, as well as site security.

Command Control Network

The CONVEX CCN is a two-way radio system using tone alert signaling to provide communications among the control room, the CONVEX load dispatcher and other key NU operating facilities.

This system is controlled by the radio console in Units 1, 2, and 3. The transmitter/receiver base station is installed in the Condensate Polishing Facility (CPF). It is installed in an impact-resistant, 41-inch cabinet bonded to electrical ground. AC voltage is the primary power source. The base station is fully solid-state incorporating integrated circuitry, located on plug-in modules or independent printed circuit boards. Highly reliable reed switches are used for antenna switching. The base station produces 98-174 (98-155)

13.8 V DC to supply power; and draws little current. Unheated, temperature compensated plug-in oscillator modules are used for frequency control. The unit contains a continuous duty transmitter that can operate indefinitely on full power. There are five front-mounted metering receptacles for ease of maintenance troubleshooting. The station is remotely controlled by tone frequencies. The wire line controlling the station need not have DC continuity for operation.

The base station is connected to the antenna via a jacketed 1/2-inch diameter semirigid coaxial cable. The cable is installed in cable tray OTX 850N which is dedicated to communication cables only. The cable ultimately terminates at the antenna mount on the CPF penthouse. The coaxial cable has the outer copper jacket bonded to ground before entry into the building. The coaxial cable has an impedance of 50 ohms and offers a combination of remarkable flexibility, high strength, and superior electrical performance. It includes a copper clad aluminum center conductor, low loss cellular polyethylene foam dielectric, corrugated copper outer conductor, and a protective black polyethylene jacket. The antenna is rigidly mounted to a permanent bracket secured to the parapet of the CPF penthouse. It is a highly directional r-f radiating device with a power gain of 5 dB. The antenna is designed to withstand severe environmental conditions. Radiating elements are made of 3/4-inch diameter tubing and reinforced with 7/8-inch diameter sockets at the mounting boom. It contains direct ground lightning protection and has a wind rating survival of 97 mph. The installed antenna weighs 37 pounds. (98-155) (98-155)

Waterford Police Radio

The Waterford Police Department two-way radio system provides communications between the Waterford Emergency Communications Dispatcher and the Control Room. The system is controlled by the radio console in Units 1, 2 and 3. The base station is located in the CPF telecommunications room. (98-155)

The antenna is installed on the CPF Building penthouse. It is provided with lightning protection and has a wind rating of 150 mph.

Tri-Town UHF Radio System

The Tri-Town UHF radio system is an administrative two-way radio system used by three towns in the Millstone area. Each of these towns has the ability to call the control room using tone alert signaling.

The system is controlled by the consoles in Units 1, 2, and 3 and base/control station and repeater relay station. The base/control station is located in the CPF Building. It contains two transmit frequencies, the second frequency being "talk-around" in the event of a repeater relay station failure. The unit is installed in an impact resistant cabinet bonded to the electrical ground. The primary power source is 120 V AC with dual 13.8 V DC battery backup. The station is fully solid-state. The station is (98-155)

connected to the antenna via a 1/2-inch jacketed semirigid coaxial cable. The cable is 20 feet in length and is securely clamped to the building bulkhead with stainless steel clamps. The cable consists of a copper clad center conductor surrounded by a low loss foam dielectric. A corrugated copper outer conductor encloses this and the entire cable is jacketed with black polyethylene. The antenna is rigidly mounted to the building exterior wall. The antenna is a heavy duty, lightweight, two-stack collinear array designed to provide 5 dB of gain, broad bandwidth, and minimum pattern distortion. A binary cable harness is used to ensure equal in-phase power distribution to all radiating elements. The wind survival of the antenna is 125 mph, and all elements are operated at DC ground to ensure immunity from lightning damage. (98-155) (98-155)

The repeater relay station is located in the Telecommunications Radio Housing adjacent to the Millstone stack. The repeater is fully solid-state and has r-f control capabilities to turn the unit on and off. The cabinet is bonded to electrical ground and its primary power source is 120 V AC backed up by the security diesel. (98-155) (98-155)

The station is connected to the antenna via a 7/8-inch jacket semirigid coaxial cable. The cable length is 150 feet and is securely clamped to the stack with stainless steel clamps. The cable consists of a copper clad center conductor surrounded by a low loss foam dielectric. A corrugated copper outer conductor encloses this and the entire cable is jacketed with black polyethylene. The antenna is rigidly mounted to the stack exterior wall. The antenna is a heavy duty, lightweight two-stack collinear array designed to provide 9 dB of gain, broad bandwidth, and minimum pattern distortion. A binary cable harness is used to ensure equal in-phase power distribution to all radiating elements. The wind survival of the antenna is over 125 mph, and all elements are operated at DC ground to ensure immunity from lightning damage.

State Police Radio System

The State Police radio system uses two frequencies. One frequency is used for radio tests and short duration communications. The second frequency is used for communications over extended periods of time. Tone alert signaling is used to allow State Police calls to the control room.

The system is controlled by the consoles in Units 1, 2, and 3. The base station is a desk top style and is located in the CPF Building 212 telecommunications room. The station fully utilizes the advantages of solid-state circuits; reliability, small size, ruggedness, and low maintenance requirements. Efficient heat radiators ensure safe operating temperatures for the transmitter power amplifier stages, and the power supply regulator transistors. The stations primary power source is 120 V AC, and it is protected from overcurrent conditions. (94-4)

The base station is connected to the antenna via a jacketed 1/2-inch diameter semirigid coaxial cable. The cable is installed in cable tray

OTX 850N which is dedicated to communication cables only. The cable ultimately terminates at the antenna mount on the CPF Building 212 penthouse. The coaxial cable has the outer copper jacket bonded to ground before entry into the building. The coaxial cable has an impedance of 50 ohms, and offers a combination of remarkable flexibility, high strength, and superior electrical performance. It includes a copper clad aluminum center conductor, low loss cellular polyethylene foam dielectric, corrugated copper outer conductor, and protective black polyethylene jacket. The antenna is rigidly mounted to permanent brackets secured to the parapet of the CPF Building 212 penthouse. The antenna is a unity power gain omnidirectional antenna with a wind rating survival of 100 miles per hour. The antenna uses a shunt-fed coaxial design in a rugged two piece construction. The lower section is enclosed in a heavy wall fiberglass tube, and the upper fiberglass whip fastens via a protected 1/2-inch by 20 thread connector. The antenna has direct ground lightning protection, and requires no ground plane elements for proper radiation. The antenna weighs 10 pounds. (94-4) (94-4)

VHF Radio Paging System/Emergency Notification and Response System (98-155)

Station Emergency Response Organization (SERO) personnel are notified via the Alpha Numeric VHF paging system. (98-155)

The Emergency Notification & Response System (ENRS) is a high-performance radio paging, voice store/forward, and message delivery system designed to meet the communications needs of nuclear power facilities, including specifically the requirements of notification outlined in 10CFR, Part 50, Appendix E, and NUREG 0654, Appendix E. (3/91) (98-155)

In the event of a nuclear accident, the ENRS allows station personnel to radio page individuals or groups of individuals with simple commands, then conveniently set up and record voice messages for retrieval by the paged individuals. As page recipients call in, they are automatically identified to the system so that station personnel can continuously monitor who has, and who has not, responded to the page, and conveniently view information on alternative means of contacting personnel (e.g., telephone numbers, etc.). (98-155)

The ENRS, as equipped, will also provide textual notification of specified organizations. (98-155)

Four of the paging transmitters are controlled via the microwave system, and one is controlled by critical repair, dedicated phone circuits.

Each paging transmitter uses 4-wire tone control for activation, and is connected to the antenna via a 7/8-inch jacketed semirigid coaxial cable. The antennas are 5.25 dB gain with true omnidirectional patterns. They are of extra rugged design with specially strengthened housings and support pipes to survive in very severe weather environments. The wind survival is (3/91)

175 mph and it is direct ground lightning protected. The antenna weight is 53 pounds.

f. The carrier current system provides direct communications with the substations at the termination of each of the outgoing transmission lines. (98-69)

g. The 5-channel amplified system is used for maintenance purposes such as instrument calibration, equipment adjustment, and the like. The layout provides point-to-point service; as between the control room and a station within the unit, or between two stations within the unit. The instruments are not permanently installed, but are the portable type that can be plugged into jacks conveniently located throughout the unit. This jack system covers working areas, the main control board, and the operator's desk. (98-69)

h. The fuel handling system telephones are located along the fuel transfer canal, and provide ready communication among those engaged in loading or unloading fuel. By means of coupling units to the crane power supply, a carrier system ties in a telephone on the polar crane and on the spent fuel cask crane. All stations are "common talk," and ringing facilities are not included. (98-69)

i. The public address facilities consist of a voice-paging system that provides communication for the Unit 2 area. A switch is provided so that the unit operator can mute all outdoor speakers at night. A paging adapter is furnished by the telephone company so that designated PBX stations under b. above can dial into the paging system of Unit 1, Unit 2, Unit 3, or all. (98-69)

j. In the unlikely event that all personnel must evacuate the area, switches in the control room energize containment, site and plant evacuation alarms. A distinctive tone generated by audio frequency oscillators is broadcast through the public address system loudspeakers. This signal takes precedence over all other use of the paging system. (98-69)

7.8.4 Availability and Reliability

7.8.4.1 Special Features

"Walkie-talkie" radios are available for communications between the reactor polar crane and the operating floor of the containment structure. They are also available for other intra-plant uses.

Administrative procedures prevent hand-held UHF radios from affecting the solid state reactor protection and/or Engineered Safety Features (ESF) systems. (3/91)

The cables in the communication systems are independent from those of other systems and are shielded or isolated from power cables and any other sources of line noise which could adversely affect the audibility of the systems. The communication systems use twisted, balanced audio pairs to further reduce the effects of longitudinally induced magnetic noise.

7.8.4.2 Design Evaluation

The failure of any system does not cause the malfunction of the other systems. To ensure high power supply reliability, nonvital systems (requiring power) receive power from the 120/208 V nonvital bus (Section 8.3.1), the Technical Support Center (TSC) electrical distribution system, or the normal DC power system (Section 8.3.2). The plant-switched network is provided with a backup power system is equipped with a rectifier and backup battery. The NU microwave system is provided with a separate battery-rectifier power system. The normal and emergency power supply systems for the SNETCO message network are located at the telephone company operating facilities. (3/91)

7.8.4.3 Testing and Inspection

The design of the communication systems permits routine testing and inspection without disrupting normal communication facilities. The evacuation alarm system will be tested periodically in accordance with normal station procedure.

TABLE 7.5-3

REGULATORY GUIDE 1.97 (REV. 2)
ACCIDENT MONITORING INSTRUMENTATION

RG 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
A-01	Pressurizer Pressure LR: P-103, 103-1 HR: P-102A, B	LR: 0-1600 PSIA HR: 1500-2500 PSIA	
A-02	Deleted		
A-03	RCS Hot Leg Temperature T-111X, T-121X	150-750°F	
A-04	Steam Generator Pressure P-1013A, B P-1023A, B	0-1000 PSIA	
A-05	Steam Generator Level L-1113A, B L-1123A, B	0-100% (top of tube bundles to separators)	
	L-1114A, B L-1124A, B	0-460" (20" above tube sheet to top of moisture separators)	
A-06	Deleted		
A-07	Containment Hydrogen Concentration AE-8152, 8154	0-10%	(99-40)
A-08	Refueling Water Storage Tank Level L-3001, L-3002	4 to 100%	
A-09	RCS Cold Leg Temperature T-115 & T-125	0°-750°F	
B-01	Neutron Flux WR-LOG-A, D (DRWR) JI-001, JI-004 (MCB)	 10^{-8} to 150% FP 10^{-8} to 100% FP	
B-02	Control Rod Position	Full in or not full in	
B-03	RCS Soluble Boron Concentration PASS	0-6000 PPM Pass	See Note 1
B-04	RCS Cold Leg Temperature T-115 & T-125	0°-750°F	
B-05	RCS Hot Leg Temperature T-111X & T-121X	150°-750°F	
B-06	RCS Cold Leg Temperature T-115 & T-125	0°-750°F	
B-07A	RCS Pressure LR: P-103, 103-1 HR: P-102A, B	LR: 0-1600 PSIA HR: 1500-2500 PSIA	

TABLE 7.5-3

REGULATORY GUIDE 1.97 (REV. 2)
ACCIDENT MONITORING INSTRUMENTATION

RG 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
B-07B	Pressurizer Pressure (Wide Range) P-102B-1	0-3000 PSIG	
B-08	Core Exit Temperature T-10 through T-450 (ICI)	200-2300°F	
B-09	Coolant Level in Reactor HJTC-A, B	Top of core to top of vessel	
B-10	Degree of Subcooling ICCM Z1 & Z2	200°F Subcooling to 35°F Superheating	
B-11A	RCS Pressure LR: P-103, 103-1 HR: P-102A, B	LR: 0-1600 PSIA HR: 1500-2500 PSIA	
B-11B	Pressurizer Pressure (Wide Range) P-102B-1	0-3000 PSIG	
B-12A	Containment Sump Water Level (Wide Range) L-8242, 8243	0' to 7' 0 to 565,000 gallons	See Note 2
B-12B	Containment Sump Water Level (Narrow Range) L-9155 (L-9155A Backup)	0-100%	See Note 3
B-13	Containment Pressure P-8113, 8114	0-60 PSIG	(99-40)
B-14	Containment Isolation Valve Position ZS-198, 505, 506, 516, 1060, 1062, 1064, 2525, 4246, 4248, 4250, 4251, 7311, 7312, 7690, 8121, 8122, 8124, 8150, 8151, 8377, 8378, 8379, 8380, 8656, 9015, 9016, 9125, 9126, 9150, 9151, 9230	Closed-Not closed	
B-15	Containment Pressure P-8238 & 8239	0-250 PSIA	
C-01	Core Exit Temperature T-10 through T-450 (ICI)	200°-2300°F	
C-02	Radioactivity Concentration or Radiation Level in Circulating Primary Coolant Pass	None PASS	See Note 1
C-03	Analysis of Primary Coolant (Gamma Spectrum) PASS	None PASS	See Note 5

TABLE 7.5-3

REGULATORY GUIDE 1.97 (REV. 2)
ACCIDENT MONITORING INSTRUMENTATION

RG 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS	
C-04A	RCS Pressure LR: P-103, 103-1 HR: P-102A, B	LR: 0-1600 PSIA HR: 1500-2500 PSIA		
C-04B	Pressurizer Pressure (Wide Range) P-102B-1)	0-3000 PSIG		
C-05	Containment Pressure P-8238 & 8239	0-250 PSIA		
C-06A	Containment Sump Water Level (Wide Range) L-8242, 8243	0′ to 7′ 0 to 565,000 gallons	See Note 2	(99-40)
C-06B	Containment Sump Water Level (Narrow Range) L-9155 (L-9155A Backup)	0-100%	See Note 3	
C-07	Containment Area Radiation RM-8240 & 8241	10^0 to 10^8 R/HR		
C-08	Effluent Radioactivity-Noble Gas Effluent from Condenser Air Removal System Exhaust RM-5099, RR-9373	10^{-6} μCi/cc to 10^{-2} μCi/cc		
C-09A	RCS Pressure LR: P-103, 103-1 HR: P-102A, B	LR: 0-1600 PSIA HR: 1500-2500 PSIA		
C-09B	Pressurizer Pressure (Wide Range) P-102B-1	0-3000 PSIG		
C-10	Containment Hydrogen Concentration AE-8152, 8154	0-10%		99-68
C-11	Containment Pressure P-8238, 8239	0-250 PSIA		
C-12	Containment Effluent Radioactivity-Noble Gases from Identified Release Points-see Variable C-14	See Variable C-14		
C-13	Radiation Exposure Rate (inside buildings or areas, which are in direct contact with primary containment where penetrations and hatches are located) None	Deleted in R.G. 1.97, Rev. 3	See Note 6	

TABLE 7.5-3

REGULATORY GUIDE 1.97 (REV. 2)
ACCIDENT MONITORING INSTRUMENTATION

RG 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
C-14	Effluent Radioactivity-Noble Gases RM-8168 RM-8132B (RM-1705-79 - Unit 1)	6×10^{-7} to 2.3×10^{5} μCi/cc	
D-01	RHR System Flow F-306	0-7000 GPM	
D-02	RHR Heat Exchanger Outlet T-303X, Y T-351Y	0-400°F	
D-03A	Accumulator Tank Level L-311, 321, 331, 341	0-100%	
D-03B	Accumulator Tank Pressure P-311, 321, 331, 341	0-250 PSIG	
D-04	Accumulator Isolation Valve Position Z-614, 624, 634, 644	Closed or open	
D-05	Boric Acid Charging Flow F-212	0-140 GPM	
D-06	Flow in HPSI System F-311, 321, 331, 341 (Backup variable: Pump Motor Current)	0-300 GPM	(99-40)
D-07	Flow in LPSI System F-312, 322, 332 & 342 (Backup variable: Pump Motor Current)	0-2000 GPM	
D-08	Refueling Water Storage Tank Level L-3001, 3002	4-100%	
D-09	Reactor Coolant Pump Status P40A, B, C, D	0-600 AMPS	
D-10	Primary System Safety Relief Valve Positions Z-200, 201, 402, 404	Closed-not closed	
D-11	Pressurizer Level L-110X, Y	0-100%	
D-12	Pressurizer Heater Status (Proportional) L105 (AM-B0504), 106 (AM-B0609)	0-250 AMPS	
	Pressurizer Heater Status (Backup) L101, 102, 103, 104 (Lights)	On-off	

TABLE 7.5-3

REGULATORY GUIDE 1.97 (REV. 2)
ACCIDENT MONITORING INSTRUMENTATION

RG 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
D-13	Quench Tank Level L-116	0-100%	
D-14	Quench Tank Temperature T-116	0-300°F	
D-15	Quench Tank Pressure P-116	0-100 PSIG	
D-16	Steam Generator Level L-1113A, B L-1123A, B	0-100% Top of tube bundles to separators	
	L-1114A, B L-1124A, B	0-460" 20" above tube sheet to separator	
D-17	Steam Generator Pressure P-4223 & 4224	0-1200 PSIA	
D-18	SRV Position FS-4225, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 38, 39, 40	Open-closed	
D-19	Main Feedwater Flow F-5268A, B F-5269A, B	0-63 x 10^6 lbs/hr	(99-40)
D-20	Auxiliary Feedwater Flow F-5277A, B F-5278A, B	0-600 GPM	
D-21	Condensate Storage Tank Level L-5282, LIS-5489, L-5280	0-100%	
D-22	Containment Spray Flow F-3023, 3024 (Backup variable: Pump Motor Current)	0-5000 GPM	
D-23	Heat Removal by Containment Fan Heat Removal System T-6082, 6086, 6090, 6093 (Backup variables: Containment Outlet Temp. & Flow) T-6031, 6032, 6033; F-6081, 6085, 6089, 6094)	0-200°F	
D-24	Containment Atmosphere Temperature T-8095, 8096, 8097, 8098, 8108, 8109 & 8110	0-350°F	
D-25	Containment Sump Water Temperature None		See Note 7

TABLE 7.5-3

REGULATORY GUIDE 1.97 (REV. 2)
ACCIDENT MONITORING INSTRUMENTATION

RG 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
D-26	Make Up Flow-In (Charging) F-212	0-140 GPM	
D-27	Letdown Flow-Out F-202	0-140 GPM	
D-28	Volume Control Tank Level L-226	0-100%	
D-29	Component Cooling Water Temperature to ESF System T-6031, 6032, 6033	0-200°F	
D-30	Component Cooling Water Flow to ESF System F-6034, 6035	0-10,000 GPM	
D-31	High Level Radioactive Liquid Tank Level (PDT) L-9051	0-100%	
D-32	Radioactive Gas Holdup Tank Pressure P-9128	0-25 PSIG	
D-33	Emergency Ventilation Damper Position ZS-8000, 8001, 8002, 8003A, 8003B, 8003C, 8004, 8005, 8006, 8007, 8009, 8010, 8361, 9506, 9507, 9508	Open-closed	See Note 8
D-34	Status of Standby Power and Other Energy Sources Important to Safety (Hydraulic, Pneumatic) Various	Volts, Amps, Pressures	See Note 9
E-01	Containment Area Radiation (High Range) RM-8240, 8241	10^0 to 10^8 R/HR	
E-02	Radiation Exp. Rate Rm-7890, 7891, 7892, 7894, 7895, 7896, 7897, 7899, 8139, 8142, 8156, 8157	1×10^{-1} to 10^4 mR/hr	
E-03A	Common Vent-Noble Gas See Variable C-14	See Variable C-14	
E-03B	Plant Vent Flow MP2: F-8412 MP1: F-20-34	10 to 10^5 CFM (MP2) 0 to 223,000 CFM (MP1)	

(99-40)

TABLE 7.5-3

REGULATORY GUIDE 1.97 (REV. 2)
ACCIDENT MONITORING INSTRUMENTATION

RG 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
E-03C	Vent from Steam Generator or Steam Dump RM-4299A, B & C	10^{-1} μCi/cc to 10^{3} μCi/cc	
E-04	(Particulates and Halogens) All Identified Plant Release Points RM-8132A/B RM-8168 RM-1705-79	Sampler Particulate & Iodine filters are used for laboratory analysis 1×10^{-3} to 1×10^{2} μCi/cc	See Note 11
E-05A	Radiation Exposure Meters (Continuous indication at fixed locations) None	None	See Note 6
E-05B	Airborne Radio-Halogens and Particulates None	None	See Note 12
E-05C	Plant and Environs Radiation None	None	See Note 12
E-05D	Plant and Environs Radioactivity None	None	See Note 15
E-06	Wind Direction Speed Temperature Various	0-360° 0-100 mph (-10)-(+18) °F	See Note 13
E-07	Accident Sampling Capability (Analysis Capability on Site) PASS	As specified in RG 1.97 Rev. 2	See Note 14

(99-40)

TABLE 7.5-3

REGULATORY GUIDE 1.97 (REV. 2)
ACCIDENT MONITORING INSTRUMENTATION

NOTES

NOTE (1) This variable is monitored with the Post Accident Sampling System (PASS) which is installed in accordance with NUREG-0737, item II.B.3.

NOTE (2) During normal operation, containment narrow range sump level is used to indicate level and initiate an alarm for the operator to manually start the sump pumps. Because the narrow range sump would be filled to capacity in the event of an accident, following an accident the containment wide range sump level would be used to monitor containment water level.

NOTE (3) Containment narrow range sump level, transmitter LT-9155A is an installed spare that is not connected to any indicator. It is a backup that can be easily terminated to replace the normal narrow range sump level transmitter LT-9155.

NOTE (4) Deleted.

NOTE (5) No existing instrument monitors this variable. Post Accident Sampling System (PASS) which is installed in accordance with NUREG-0737, item II.B.3 will be utilized.

NOTE (6) Deleted on R.G. 1.97, Rev. 3.
Also, not cost-effective per NUREG/CR 2644.

NOTE (7) No existing instrument monitors this variable.

NOTE (8) The following is a list of dampers and limit switches for variable D-33:

2-HV-206B	2-HV-206A	2-HV-207	2-HV-201	2-HV-208	2-HV-209	2-HV-211
ZS-8000	ZS-8001	ZS-8002	ZS-8003A	ZS-8003B	ZS-8003C	ZS-8004
2-HV-210	2-HV-212A	2-HV-212B	2-HV-203A	2-HV-203B	2-HV-202	
ZS-8005	ZS-8006	ZS-8007	ZS-8009	ZS-8010	ZS-8361	

NOTE (9) The status is indicated by voltmeters, ammeters, watt meters, and status lights on the main control board. The status of the starting air for the diesels is alarmed in the control room. All sensors are located in a mild environment.

NOTE (10) Deleted.

NOTE (11) Particulate and iodine filters for these monitors are removed for laboratory analysis.

NOTE (12) Portable instruments are used to monitor this variable per R.G. 1.97, Rev. 2.

NOTE (13) Actual wind direction indicates a 0°-540° range (1.5 revolutions) for computer averaging purposes.

NOTE (14) The (PASS) will be utilized for this variable. Total Dissolved Gas (TDG) capability has been eliminated by DCR M2-98098, Rev. 0.

NOTE (15) Isotopic analysis via various on-site and off-site gamma (GeLi) spectrometers.

TABLE 7.5-6

AIRBORNE PROCESS/EFFLUENT RADIATION MONITORS

CHANNEL NO.	DESCRIPTION	DETECTOR TYPE	SAMPLER TYPE	DECADE RANGE
RM-8132* A	Unit 2 Stack Monitor Particulate	B-Scint.	Off-Line Part.	10 - 10^6 cpm
RM-8132 B	Unit 2 Stack Monitor Gaseous	B-Scint.	Off-Line Gas	10 - 10^6 cpm 99-86
RM-8168	Unit 2 Stack Mid and High Range	GM Tubes	Off-Line Gas & Part.	10 x E-3 to 10 x E+5 μCi/cc
RM-8123* A	Containment Air Monitor - Particulate	B-Scint.	Off-Line Part.	10 - 10^6 cpm
RM-8123 B	Containment Air Monitor - Gaseous	B-Scint.	Off-Line Gas	10 - 10^6 cpm
RM-8262* A	Containment Air Monitor - Particulate	B-Scint.	Off-Line Part.	10 - 10^6 cpm
RM-8262 B	Containment Recirculating Air Monitor - Gaseous	B-Scint.	Off-Line Gas	10 - 10^6 cpm
RM-8434* A	Radwaste Vent Monitor - Particulate	B-Scint.	Off-Line Part.	10 - 10^6 cpm
RM-8434 B	Radwaste Vent Monitor - Gaseous	GM Tube	Off-Line Gas	10 - 10^6 cpm
RM-5099	Condenser Air Ejector Discharge Monitor - Gaseous	γ-Scint.	Adjacent to Line Gas	10 - 10^7 cpm

TABLE 7.5-6

AIRBORNE PROCESS/EFFLUENT RADIATION MONITORS

CHANNEL NO.	DESCRIPTION	DETECTOR TYPE	SAMPLER TYPE	DECADE RANGE	
RM-8011	Control Room Monitor - Gaseous	GM Tube	Off-Line Gas	10 - 10^6 cpm	
RM-9799 A & B	Control Room Intake Duct	GM Tube	On-Line Gas	0.1 - 10^4 mr/hr	
RM-8145* A	Fuel Handling Exhaust Air Monitor - Particulate	B-Scint.	Off-Line Part.	10 - 10^6 cpm	
RM-8145 B	Fuel Handling Exhaust Air Monitor - Gaseous	GM Tube	Off-Line Gas	10 - 10^6 cpm	(6/90)
RM-8997*	Radwaste Vent Monitor - Particulate	B-Scint.	Off-Line Part.	10 - 10^6 cpm	
RM-8998*	Radwaste Vent Monitor - Particulate	B-Scint.	Off-Line Part.	10 - 10^6 cpm	
RM 8999*	Radwaste Vent Monitor - Particulate	B-Scint.	Off-Line Part.	10 - 10^6 cpm	
RM-9095	Filtered Waste Gas to Unit 1 Stack Monitor - Gaseous	B-Scint.	Off-Line Gas	10 - 10^6 cpm	(98-37)

*Replaceable charcoal cartridge assembly is provided for laboratory analysis of iodine.

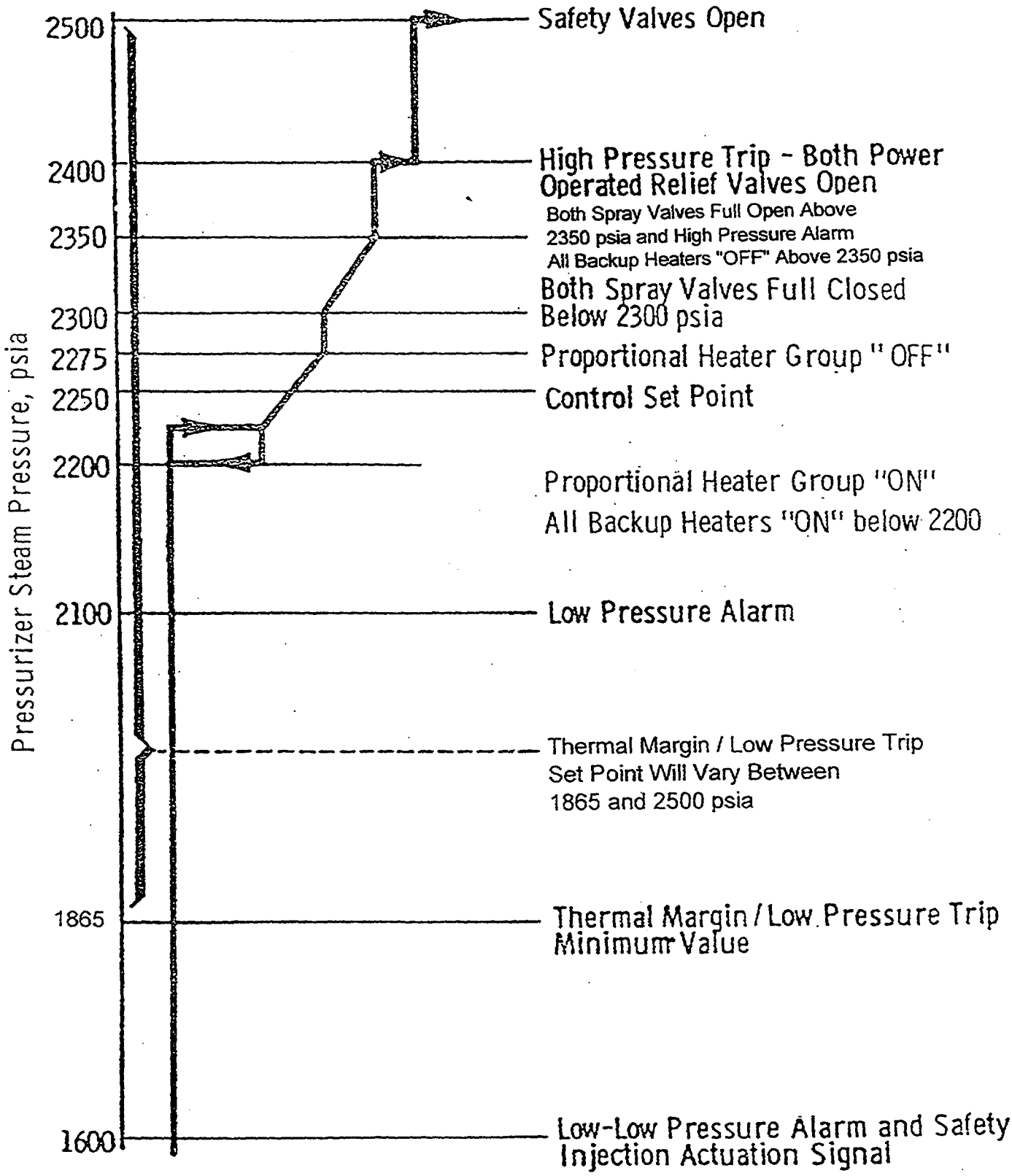

FIGURE 7.4-4
PRESSURE CONTROL PROGRAM

Rating	125-volts DC/120-volts AC, 15 kVA, any pf, 2 wire, ungrounded
Voltage regulation (%)	±1
Frequency stability (%)	±0.5
Overload capability (%)	125% of continuous rating for 5 minutes at any power factor
Transfer time of static switch	Zero break

A common inverter trouble alarm for each Inverter and Static Switch is provided in the Control Room. The following alarms with local alarm lights, except as noted, are tied to the common inverter trouble alarm:

- Synchronism Fail
- Input CB Tripped
- Output Undervoltage
- Alternate Line (light on static switch)
- AC Ground Fault
- Over Temperature (inverters only) 99-87
- Reverse Polarity
- Low Air Flow (static switches only/no local alarm light)
- Static Switch Bypassed (no local alarm light) 99-87

Active and passive components of the reactor protective system, engineered safety feature circuits and the emergency power system are designed to meet the seismic design requirements. The equipment and their components are designed to function before, during, and after an operating basis earthquake. In addition, they can sustain a design basis earthquake without any loss of protective function.

Seismic qualification of the Inverter and Static Switches was demonstrated by Wyle Laboratories in their Test Report No. 42384-1: "Seismic Simulation Test Program On A 15 kVA Inverter/Static Switch Assembly for use in Northeast Utilities' Millstone Unit 2 Nuclear Plant," dated February 13, 1992. The details of qualification are also documented in the Seismic Qualification Review 92-034: DC/AC Inverter Replacement.

Seismic qualification tests were performed for five (5) Operational Basis Earthquakes (OBE's) and one (1) Safe Shutdown Earthquake (SSE). The testing was performed to industry standard IEEE 344-75, which exceeds original qualification requirements for MP2. These were described as qualification to IEEE 344-71, supplemented by the requirement that simultaneous horizontal and vertical motions were required during testing. As such, the seismic qualification testing for the Inverter/Static Switch assembly was successfully performed in accordance with plant design requirements.

The distribution panels provide feeder protection by the use of molded case circuit breakers with thermal-magnetic trips. Three panelboards were tested under simulated seismic conditions; two of the well-mounted type and one floor mounted. These panelboards are

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9.9.2.3 System Operation

The purge system is not operating and the containment isolation valves are locked closed in Modes 1 through 4 by shutting them, pulling their control power fuse and locking their associated fuse block. By locking them closed in this manner, these valves are considered sealed closed isolation valves. When access to the containment is desired and the Plant is in the cold shutdown or the refueling mode, the purge fan, if required, is started and the isolation valve opened. If a high containment radiation level is detected, the containment air purge supply and exhaust valves, 2-AC-4, 2-AC-5, 2-AC-6 and 2-AC-7, receive a signal to close (FSAR Subsection 7.3.2.2.c). 99-74

9.9.2.3.1 Normal Operation

The purge system is initiated manually in the control room by operator action. Hand switches are provided to start the fan and open isolation valves and dampers. The third main exhaust fan (Section 9.9.9) is started to exhaust the containment through particulate and high efficiency particulate air (HEPA) filters (Subsection 9.9.9.2.2). The system performance is monitored by containment temperature indication (Section 7.5). Air is supplied in the containment above the operating floor. Mixing is provided by the containment air recirculation and cooling system (Section 6.5) throughout the lower elevations of the containment. Mixing is provided throughout the upper regions of the containment by the containment auxiliary circulation system (Section 9.9.3). The containment ambient conditions during shutdown are described in Subsection 6.5.3.4. (99-52)

Outside air is filtered prior to distribution within the containment. A steam heating coil is provided for tempering the outside air when required. During summer conditions, the outside air is supplied at approximately 86°F. During winter conditions, the outside air is tempered and supplied at 70°F. (99-52)

Radioactive effluents released to the site boundary resulting from the containment purge operations are described in the Environmental Report.

The containment isolation valves on the purge system are locked closed and electrically disconnected during all modes of operation except cold shutdown and refueling. However, following a Containment Isolation Actuation Signal (CIAS) the supply fan, if operating, and associated isolation damper, if open, are isolated. (99-52)

Damper closing is monitored by position indication in the control room.

The enclosure building is heated or ventilated when required by the purge system. Branch ducting is provided from the main purge line for the enclosure building. Ventilation may be required for this area periodically throughout the summer to maintain the enclosure building temperature below 120°F. Auxiliary heating, in addition to unit heater, may be required periodically through the summer to the winter to maintain the enclosure building temperature above 70°F. Unit heaters are located near Elevation 48-6 to maintain the lower regions of the enclosure building at 70°F. The upper regions in the enclosure building are

The Millstone Units 1 and 2 fire pumphouses house three 2,000-gpm at 100 psi rated fire pumps which supply the yard loops; two with electric-motor drives and one with diesel-engine drive. The Millstone Unit 1 pumphouse contains one electric driven pump (M7-8), fed from Millstone Unit 1 power, and the diesel-driven fire pump (M7-7). The Millstone Unit 2 pumphouse contains one electric driven pump (P-82) fed from Millstone Unit 2 power. All three pumps have individual connections to the underground supply system. Maximum system flow and pressure requirements can be met with any one of the three pumps out of service. The Millstone Unit 1 pumphouse also contains the jockey pump.

System operation is such that a Millstone Unit 1 50 gpm electric jockey pump (M7-11) maintains system pressure by automatically starting when line pressure drops to 105 psig and will run until pressure reaches 120 psig as indicated by a line pressure switch. A hydro-pneumatic tank is provided in the system to prevent short cycling of the jockey pump. The Millstone Unit 2 electric driven fire pump (P-82) is driven by an AC motor from the 480V Load Center. This pump is activated by a single pressure switch set at 85 psig. In the event this switch or pump fails to operate and line pressure continues to drop, the diesel-driven fire pump is activated by an additional pressure switch set at 75 psig and powered by the diesel battery system. The diesel pump is started by its own self-contained battery system. A battery charger is provided for recharging. Both the electric and diesel-driven fire pumps deliver 2000 pgm at 100 psi discharge pressure and remain in operation until they are manually shut down. Electrical interlocks stop the jockey pump when either of the two fire pumps start.

The fire pumps are supplied from two 245,000-gallon ground level suction tanks. The tanks are automatically filled through a water line fed from city water. 99-84

If a major fire in any location of the Millstone Unit 2 site should occur, the combined water tank and makeup water capacity would provide an adequate water supply for Millstone Unit 2. The necessary pressure and flow would be maintained through the use of any two simultaneously operating 2,000 gpm rated pumps.

The site fire water system can be connected to the auxiliary feedwater system to provide an alternate source of water in the event that the primary source of water in the condensate storage tank is depleted and cannot be adequately replenished. The combination of the two storage tanks with the potential for replenishment from the city water supply provide multiple, reliable sources of water with which to feed the steam generators and remove decay heat over an indefinite period of time. The Technical Requirements Manual ensures the availability of fire water.

In case of a loss of off-site power and service water, fire water may be provided for cooling one EDG via a cross-connection to the service water supply to the EDGs.

Unit 2 Appendix R analysis uses different fire area divisions based on separation requirements. The safe shutdown evaluation relies only on fire areas to determine the effects of fire on safe shutdown. These divisions result in large fire areas that do not permit accurate fire loading and equivalent fire severity analysis. Therefore, from a fire protection standpoint, the divisions in the FHA provide more indicative results with a broader range of uses.

Drawings provided in the FHA show fire zone/area division and serve as reference for this section.

9.10.3.4 Fire Hazard Analysis Results

The FHA results for each fire area are contained in the FHA (Reference 9.10-2).

9.10.4 Inspection and Testing

Administrative controls are provided through existing Plant Administrative Procedures, Operating Procedures and the Quality Services Program to ensure that the Fire Protection Program and equipment is properly maintained. This includes Quality Assurance (QA) audits of the program implementation, conduct of periodic test inspections, and remedial actions for systems and barriers out of service. This program emphasizes those elements of fire protection that are associated with safe shutdown and their significance when evaluating program and equipment deficiencies.

All fire protection equipment and systems are subject to periodic inspections and tests in accordance with the intent of National Fire Codes and the Fire Protection Program.

The technical requirements found in Millstone Unit 2 TRM (Reference 9.10-6) describe the limiting condition for operation and surveillance requirements for the FPS. These technical requirements ensure the FPS is properly maintained and operated.

Equipment out of service including fire suppression, detection, and barriers are controlled through the administrative program and appropriate remedial actions taken. The program requires all impairments to FPS to be identified and appropriate notification given to the Fire Protection System Engineer for evaluation. 99-78

As conditions warrant, remedial actions include compensatory measures to ensure an equivalent level of fire protection in addition to timely efforts to effect repairs and restore equipment to service.

9.10.5 Personnel Qualification and Testing

9.10.5.1 Fire Protection Organization

Formulation, implementation, and assessment of the effectiveness of the Fire Protection Program are delegated as indicated in Reference 9.10-1, the Fire Protection Program Manual. 99-78

The selection of these functions is based on safe shutdown performance goals identified in BTP CMEB 9.5-1 Section C.5.c(2).

The safe shutdown functions, described above, assure that the reactor will be safely shut down, cooled down, and maintained in a cold shutdown condition. The achievement of safe shutdown conditions precludes the occurrence of an unrecoverable plant condition, e.g., uncontrolled primary depressurization, loss of decay heat removal capability, or a breach of the reactor coolant system (RCS) boundaries.

9.10.6.2 Analysis of Safe Shutdown Systems and Components

The exact location, duration, and magnitude of potential fires at the plant cannot be predicated in advance. To ensure that safe shutdown can be achieved following any postulated fire, Millstone Unit 2 has been divided into fire shutdown areas. The maximum credible fire was considered for each fire shutdown area. An engineering evaluation was then conducted to ensure that a fire originating in any one area would not spread to an adjacent fire shutdown area.

All unprotected electrically operated equipment within the affected fire area was assumed to be damaged by the fire. An analysis was then done to confirm that sufficient plant systems remained operable to allow plant operators to achieve safe shutdown. Where necessary, equipment modifications were made, credit was taken for manual operation of remotely actuated valves, and operating and repair procedures were generated, to allow this goal to be achieved. By repeating this process for each fire shutdown area, a set of fire shutdown systems, components and procedures was produced. This set is adequate to ensure that safe shutdown can be achieved and maintained regardless of the location of the fire. 99-84 99-84

The above approach is considered to be very conservative. In reality, existing fire detection and suppression systems, and mitigating actions by the fire brigade, would limit the extent of fire damage. Thus, additional systems would be functional, and the operators would have additional flexibility in dealing with the effects of a fire.

Initial Assumptions

(1) The station is operating at 100% power upon the occurrence of a fire.

(2) The reactor is tripped either manually or automatically.

(3) The analysis assumes both a loss of offsite power and availability of offsite power, whichever poses the more severe shutdown condition.

(4) No additional single failure is considered other than the loss of off-site power, those failures directly attributable to the fire, and spurious operations that can be postulated to occur as a result of the fire.

(5) No component or system required for safe shutdown is assumed to be out-of-service.

(3) For Common Enclosure Concern

By utilizing the fire area approach, all shutdown equipment and associated cables within a fire area had to be identified and considered subject to a fire-induced failure, unless otherwise justified.

9.10.6.4 Reactor Inventory Isolation and Hi/Low Pressure Interfaces

Those systems that penetrate the RCS that are open to other systems and could cause diversion or loss of reactor inventory must be able to be isolated and remain isolated. Isolation is essential for those fire areas where fire could result in the loss of short-term reactor inventory make-up capability.

The systems which interface with the RCS and could divert inventory are as follows:

- Chemical and Volume Control Charging Lines
- Chemical and Volume Control Letdown Line
- Pressurizer Vent
- Pressurizer Power Operated Relief Valve (PORV) Lines
- Shutdown Cooling System
- Pressurizer Safety Valve Lines
- Auxiliary Spray Line
- Primary Drain Header
- Reactor Head Vent
- Low Pressure Safety Injection (LPSI)
- Sampling System

Some systems have been designed with flow check or nonreturn valves to ensure that they would not cause the diversion of reactor vessel inventory. For these systems, the valves which provide this isolation function are provided in Reference 9.10-3.

The remaining systems are not provided with these passive fail-safe means of ensuring isolation. The methods which are utilized to provide and maintain isolation of these remaining interfacing systems are also provided in Reference 9.10-3.

9.10.6.5 Exemptions from the Specific Requirements of Appendix R to 10CFR50, III.G.2, III.G.3, III.J and III.O

The following exemptions from the requirements of Appendix R have been granted:

(1) Exemption from 10CFR50 III.J requirement to provide eight hour battery powered emergency lighting in general yard areas. The exemption allows the use of security lights to illuminate outdoor areas to access the Vital Bus 14H Enclosure (formerly Emergency Bus 24F), Intake Structure and Refueling Water Storage Tank (RWST) Pipe Chase Enclosure.

(2) Exemption from 10CFR50 III.G.2 requirement to provide a three hour rated barrier in the Auxiliary Building between the Boric Acid Tank - Chemical Addition Area (R-1) and MCC B61 (R-2). The exemption allows crediting the steel environmental enclosure for MCC B61, and a water curtain installed to protect the enclosure from fires in this area, as a fire barrier between MCC B61 and the Auxiliary Building.

(3) Exemption from 10CFR50 III.G.2 requirement to provide a three hour rated barrier between redundant safe shutdown trains in the Auxiliary Feed Pump Pit. The exemption allows the use of a nonrated water tight door as a fire barrier between Motor Driven Auxiliary Feedwater Pumps (R-3) and Turbine Driven Auxiliary Feedwater Pumps (R-12). 99-69

(4) Exemption from 10CFR50 III.G.2 requirement to provide 20 foot horizontal separation (with no intervening combustibles) between redundant Auxiliary Feedwater Valves (2-FW-43A,B) located in the Turbine Building (R-3). The exemption concludes that fire damage to these valves would either cause them to fail open in their required position, or cause them to fail closed, but capable of being opened.

(5) Exemption from 10CFR50 III.G.2 criteria for separation of redundant charging pump trains located in the Charging Pump Room (R-4). The exemption credits the following for assuring the availability of at least one charging pump for a Charging Pump Room fire:

- low fire loading,
- partial height concrete walls,
- approximate 18 ft. horizontal separation between pumps,
- oil spill containment curbing between the pump cubicles,
- existing area fire detection system,
- rerouting of the "B" and "C" charging pump (Facility Z2) cables outside fire area R-4.

(6) Exemption from 10CFR50 III.G.3 requirement to provide a fixed suppression system in an area which relies upon alternative shutdown

capability, the Intake Structure (R-16). This exemption was granted on the basis that low combustible loading, limited intervening combustibles, 19 feet of horizontal separation between the redundant service water pumps and existing fire protection features would be sufficient to limit the potential for fire damage to safe shutdown equipment in this area and an alternate safe shutdown capability (independent from this area of concern) is available to power hot shutdown equipment via the Unit 1 backfeed capability.

(7) Exemption from 10CFR50 III.G.3 requirement to provide a fixed suppression system in an area which relies upon alternative shutdown capability, the East 480V Switchgear Room (R-11). This exemption was granted on the basis that low combustible loading, existing fire protection features and the close proximity of the control room which enhances early detection and manual suppression capability would be sufficient to limit the potential for fire damage to safe shutdown equipment in this area and an alternate safe shutdown capability (independent from this area of concern) is available to power hot shutdown equipment via the Unit 1 backfeed capability. 99-69

Additional considerations for granting this exemption related to the potential water suppression system effects on electrical components, and gaseous suppression system effects on control room habitability.

(8) Exemption from 10CFR50 III.O requirement to provide a Reactor Coolant Pump Oil Collection System designed to withstand the Safe Shutdown Earthquake (SSE). This exemption was granted on the basis of the following:

- The lubricating oil systems for the reactor coolant pumps are qualified to remain functional during and after seismic events up to and including an SSE, and
- The installed oil collection system would not cause loss of operability of safety-related equipment nor would it cause a fire in the event of an SSE.

REFERENCES

9.10-1 Millstone Nuclear Power Station Fire Protection Program Manual. (98-61)

9.10-2 Fire Hazard Analysis Millstone Unit 2, latest revision.

9.10-3 Millstone Unit 2 10 CFR 50 Appendix R Compliance Review, March 1987

9.10-4 Millstone Nuclear Power Station, Station Procedure WC-7, Fire Protection Program (95-6)

9.10-5 Millstone Nuclear Power Station, Station Procedure TQ-1, Personnel Qualification and Training (98-48)

9.10-6 Millstone 2 Technical Requirements Manual

- Reset, which resets the electrical logic for returning the mode of operations back to normal.

Auto actuation of the electric-driven pumps and the auxiliary feedwater regulating valves occurs when 3 minutes, 25 seconds have elapsed since the steam generator levels dropped to ≤26.8 percent. 99-83

A manual bypass valve is provided around each air-operated AFW regulating valve, 2-FW-43A and 2-FW-43B, in the auxiliary feedwater (AF) line to each steam generator to ensure the availability of feedwater for decay heat removal should either one of the regulator valves fail in the closed position. To meet the functional requirement of providing AFW to either or both steam generators with a limiting single failure, a normally open motor-operated cross-tie valve is provided between the AF regulating valves.

In addition, the AFW regulating valves, 2-FW-43A and 2-FW-43B, are equipped with a backup air supply to provide valve closure and valve control in the event of loss of the Instrument Air System. The backup air supply is provided by high pressure air cylinders. The system is designed to operate in a harsh environment caused by the beyond-design basis event of a feedwater line break inside the turbine building coincident with a failure of the main feedwater check valve (2-FW-5A or 2-FW-5B). The Instrument Air system is non-safety related. The backup air supply is safety related.

The AFW regulating valves receive an AFAIS signal to open upon low level in either steam generator to ensure feedwater flow following a loss of normal feedwater. These valves are also designed to fail open upon a loss of air or electric power. Following a MSLB or HELB, AFW to the affected steam generator is isolated. The AFW regulating valves can be closed by the backup air supply for a MSLB inside Containment or outside Containment upstream of the Main Steam Isolation Valve (MSIV). For HELBs inside the Turbine Building, the AFW regulating valves are open and the cross-tie motor-operated valve (MOV) is environmentally qualified to close to isolate the affected steam generator. Depending on which steam generator is affected, the motor-driven pumps may need to be shut down and the turbine-driven pump started to feed the unaffected steam generator. In the event of a single failure in the power system that causes the AFW regulating valve 2-FW-43B to fail open and the MOV cross-tie valve to fail as-is, the affected steam generator will be isolated by local operator action following a MSLB.

The auxiliary feedwater system can be used to provide long-term cooling in the event of a LOCA in conjunction with the dumping of steam. The AFW pumps would initially take a suction on the condensate storage tank. If in the long term, the CST becomes depleted and cannot be replenished by normal makeup, the operators can connect the fire water system, and its two, 250,000 gallon storage tanks, to the AFW pump suctions. The fire water storage tanks can be replenished from the city water supply if necessary. See Section 14.6.5.3 for a description of long-term cooling in the event of a LOCA.

10.4.5.4 Equipment

10.4.5.4.1 Condensate Pumps

Each condensate pump is a motor-driven, multistage, vertical, canned suction type, centrifugal unit. Three 55 percent capacity pumps are installed at an elevation that allows

CHAPTER 12

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CHAPTER 12

CONDUCT OF OPERATIONS

12.1 ORGANIZATIONAL STRUCTURE

Information regarding the organizational structure is presented in Chapter 1.0, Organization, of the NUQAP (Reference 12.1-2). With the exception given below, that information is incorporated herein by reference. 99-78

The majority owners, holding 100 percent of the Millstone 2 nuclear plant, are the Northeast Utilities System (NU) subsidiaries; Connecticut Light and Power Company (CL&P) and Western Massachusetts Electric Company (WMECO).

12.1.1 Management and Technical Support Organization

Information regarding the management and technical support organization is presented in Chapter 1.0, Organization, of the NUQAP (Reference 12.1-2). That information is incorporated herein by reference. 99-78

12.1.1.1 Technical Support for Operations

The organization providing technical support for operations is described in Chapter 1.0 of the NUQAP Topical Report (Reference 12.1-2). That information is incorporated herein by reference. 99-78

12.1.1.2 Organizational Arrangement

The organizational arrangement is as described in Chapter 1.0 of the NUQAP Topical Report (Reference 12.1-2). That information is incorporated herein by reference. 99-78

12.1.2 Operating Organization

12.1.2.1 Plant Organization

The plant organization is as shown in Reference 12.1-2. 99-78

12.1.2.2 Operating Shift Crews

The minimum shift crew composition and license requirements during all modes of operation are contained in Section 6.2 of the Technical Specifications. 99-78

12.1.3 Qualification of Nuclear Plant Personnel

Education and experience requirements are established by Section 6.3 of the Technical Specifications. 99-78

REFERENCES

12.1-1 (Deleted) 99-78

12.1-2 Northeast Utilities Quality Assurance Program (NUQAP) Topical Report, Chapter I, Organization. 99-78

12.4 REVIEW AND AUDIT

A program describing the review and audit of activities important to and affecting station safety during the operational phase has been established and complies with Regulatory Guide 1.33, "Quality Assurance Program Requirements (Operation)." The program provides a system to ensure that these activities are performed in accordance with company policy, rules, and approved procedures.

12.4.1 Onsite Review

On site review is conducted by committee(s) as described in the Northeast Utilities Quality Assurance (NUQAP) Topical Report (Reference 12.4-1). 99-78

12.4.2 Independent Review

Independent review of activities affecting the unit's safety is performed by the Nuclear Safety Assessment Board (NSAB) as described in the NUQAP Topical Report. 99-78

12.4.3 Audits

The Audit Program for activities affecting safety related systems, structures, or components is as described in the NUQAP Topical Report. 99-78

12.4.4 Other Review Groups

Other review groups evaluate, on a periodic basis, the effectiveness of the units. These groups are: independent consultants, task forces, joint utility audit groups, insurance inspections, etc.

REFERENCE

12.4-1 Northeast Utilities Quality Assurance Program (NUQAP) Topical Report. 99-78

12.5 PLANT PROCEDURES

Station procedures are written on a unit and station services level controlling the specifics of station operations, including specifications; maintenance and modification; periodic test, inspection, calibration, and special processes, and plant and equipment.

12.5.1 Administrative Procedures

12.5.1.1 Conformance with Regulatory Guide 1.33

Regulatory Guide 1.33, issued February 1978, Quality Assurance Program Requirements, and ANSI 18.7/ANS 3.2 - 1976/Standard for Administrative Controls for Nuclear Power Plants, is used as guidance for the preparation of administrative and station procedures.

12.5.1.2 Preparation of Procedures

Preparation, review and approval of procedures is as described in the NUQAP Topical Report. 99-78

12.5.1.3 Procedures

Administrative control and station procedures cover a wide range of topics. Major areas are described as follows.

Standing Orders to Operating Personnel

Administrative control procedures define the authorities and responsibilities of operating personnel. The procedures specify the number of personnel with reactor operator and senior reactor operator licenses required to be on site or at the controls in all plant conditions. The administrative procedures direct the proper maintenance, review, and disposition of operating records, establish requirements for shift turnover, and establish the authority and responsibilities of the person in charge of the control room to limit access. Procedures also direct the assignment of shift personnel to duty stations, establish limits for scheduled working hours, provide methods of timely and concise feedback of operating experience to applicable plant staff, and describe methods used to verify operational activities are performed correctly. Plant procedures also provide for shift turnover such that all necessary information is properly transmitted to the oncoming shifts.

Special Orders of a Transient or Self-Cancelling Character

Night orders are issued as necessary to provide guidance to operating shifts and are of a temporary nature but will be incorporated into an administrative control if the need becomes permanent.

Special Procedures

Special procedures are prepared as necessary to support infrequently performed activities which are not to be included in the permanent list of station procedures. A special procedure can be written for any type of station procedure (i.e., maintenance, operating). The form of a special procedure is the same as the applicable type of station procedure. All requirements for review, approval, revisions, and changes are the same as for permanent station procedures.

12.5.2 Operating and Maintenance Procedures

Operating and maintenance procedures are divided into several categories which are described in the following subsections. The list of these procedures is contained in the Master Document Index.

Operating and maintenance procedures preparation is the responsibility of the appropriate department head. When a procedure is written, the applicable Department Head/Manager is responsible to forward the procedure for review and approval in accordance with the NUQAP Topical Report. 99-78

Plant operations are performed in accordance with written and approved Station and Department procedures.

Independent position verification of safety-related components/systems (valves, breakers, and control switches) with no indication in the control room are performed prior to the return-to-service of the component/system.

12.5.2.1 Control Room Operating Procedures

12.5.2.1.1 General Operating Procedures

These procedures cover major plant evolutions. Step-by-step instructions are provided for the function or task with the appropriate cross reference to system operating procedures for details of specific system operation. Appropriate precautions and limitations are included.

12.5.2.1.2 System Operating Procedures

These procedures provide step-by-step details for systems operations with appropriate prerequisites, precautions, limitations, and alarm responses. Each procedure covers the expected modes of operation of the system as well as startup, shutdown, filling and venting, and standby operation as applicable.

12.5.2.1.3 Abnormal Operating Procedures

Operating procedures are prepared for abnormal operation of the unit. Abnormal operation is a condition that could degrade into an emergency or could violate Technical Specifications if proper action were not taken. These procedures identify the symptoms of the

12.6 RECORDS

Records are kept and maintained in accordance with the applicable federal, state, and operating license requirements. The records and retention program is administered at the plant site and is in general conformance to American National Standards Institute (ANSI) N45.2.9, 1974. These controls include all quality related records including plant and as-built drawings. 99-78 99-78

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14.8 MILLSTONE UNIT 2 FSAR EVENTS NOT CONTAINED IN THE STANDARD REVIEW PLAN

14.8.1 Failures of Equipment Which Provides Joint Control/Safety Functions

Millstone Unit 2 has no instrumentation which serves a combined function of process control and of initiation of emergency safety systems. 99-65

14.8.2 Containment Analysis

14.8.2.1 Main Steam Line Break Analysis

14.8.2.1.1 Event Initiator

In the event of a Main Steam Line Break (MSLB), the release of steam into containment will result in a rise in both temperature and pressure. The break is assumed to occur in the piping between the steam generator and the containment wall penetration. Mass and energy releases are limited by the flow restrictor in the steam generator outlet nozzle.

14.8.2.1.2 Protective Systems

Engineered Safety Features (ESF) systems which will operate to terminate the mass and energy release to containment and suppress containment atmosphere temperature and pressure are the Main Steam Isolation Signal (MSIS), Safety Injection Actuation Signal (SIAS), and Containment Spray Actuation Signal (CSAS).

A MSIS will actuate on receipt of a containment high pressure signal to shut the following valves and trip the main feedwater pumps:

1.	Steam Generator 1&2 Isolation Valves	(HV-4217 & HV-4221	or MS-64A&B)
2.	Steam Generator 1&2 Isolation Valves Bypass	(HV-4218 & HV-4222	or MS-65A&B)
3.	Steam Generator 1&2 Feedwater Isolation Valves	(HV-5419 & HV-5420	or FW-5A&B)
4.	Steam Generator 1&2 Feedwater Regulating Valves	(FV-5268 & FV-5269	or FW-51A&B)
5.	Main Steam Leg Low PT. Drains	(HV-4193 & HV-4209	or MS-265B & MS-266B)
6.	Steam Generator 1&2 Feedwater Regulating Bypass Valves	(FV-5215 & FV-5216	or FW-41A&B)
7.	Feedwater Block Valve to Steam Generators 1&2	(HV-5263 & HV-5264	or FW-42A&B)
8.	Steam Generator 1&2 Feed Pump Discharge Valves	(HV-5245 & HV-5247	or FW-38A&B)

All MSIS components will trip on activation of either channel or MSIS.

A SIAS will activate the Containment Air Recirculation (CAR) fans and give a start signal to the emergency diesel generators (EDGs).

14.8.2.1.7 Conclusions

The results of a MSLB initiated from 0 percent reactor power with coincident failure of the Vital Bus Cabinet VA-10 or VA-20 produces the limiting containment peak pressure of 53.8 psig. The peak containment atmospheric temperature for this case is 377.9°F. The higher initial steam generator inventory along with the addition of maximum AFW for 10 minutes (the time assumed for operator action) is more limiting than cases initiated from higher power with MFW. The plant response for the limiting peak pressure case is shown in Figures 14.8.2-1 through 14.8.2-9 and the sequence of events is given in Table 14.8.2-4. 99-73

The results of MSLB initiated from 102 percent reactor power with coincident failure of the Vital Bus Transfer Mechanism produces the limiting containment peak atmospheric temperature of 417.67°F. 99-73

The containment pressure remains below the design pressure of 54 psig. Although peak containment atmospheric temperature exceeds the 289°F design temperature, it is only for a short period of time and does not raise the containment structure above 289°F.

14.8.2.2 Loss of Coolant Accident Analysis

14.8.2.2.1 Events Analyzed

Nine separate cases of Loss of Coolant Accidents (LOCAs) were analyzed with variations for break locations, single failures and availability of offsite power. Break locations analyzed are the reactor coolant pump suction leg, pump discharge leg and hot leg. Each break is a double-ended slot break because this had been determined to be the limiting break in previous analyses. (The term double-ended or single-ended when associated with a slot break implies that the slot-break area is equivalent to the break area for either a double-ended or single-ended guillotine break in the same reactor coolant loop.) Break designations and sizes are Double-Ended (9.82 sq. ft.) Suction Leg Slot (DESLS) Break, Double-Ended (9.82 sq ft) Discharge Leg Slot (DEDLS) Break and Double-Ended (19.2 sq ft) Hot Leg Slot (DEHLS) break. Single failures considered are failure of an emergency diesel generator (EDG) (for a loss of power (LOP) case) which fails 1 train of containment heat removal systems or (for no LOP), failure of either 1 spray system or 2 CAR fans. With an LOP 1 train of ECCS will operate, with no LOP both trains will operate.

Case 1: DESLS with an LOP, failure of an EDG, min ECCS

Case 2: DESLS without LOP, failure of 1 spray, max ECCS

Case 3: DESLS without LOP, failure of 2 CAR fans, max ECCS

Case 4: DEDLS with an LOP, failure of an EDG, min ECCS

Case 5: DEDLS without LOP, failure of 1 spray, max ECCS

Case 6: DEDLS without LOP, failure of 2 CAR fans, max ECCS

14.8.3 Hydrogen Accumulation in Containment

14.8.3.1 General

The results of an analysis of the accumulation of hydrogen in the containment following the DBA indicate that hydrogen can be safely and effectively controlled by an electric hydrogen recombiner system. This hydrogen recombiner system will be used, if necessary, to maintain the maximum hydrogen concentration at or below three (3) volume percent. As stated in Regulatory Guide 1.7, the lower flammability limit for hydrogen in air saturated with water vapor at room temperature and atmospheric pressure is assumed to be four (4) volume percent. (98-197)

Therefore, the proposed system will be capable of maintaining the hydrogen concentration at a safe nonflammable level. The hydrogen control systems are described in Section 6.6.

14.8.3.2 Method of Analysis for Hydrogen Accumulation

The significant sources of hydrogen following the DBA are radiolysis of water from the decay of fission products, the zirconium- water reaction, and corrosion of containment metals. Each of these potential sources has been considered in detail and the hydrogen generated from each has been calculated. The calculations conformed in all respects to the conservative assumptions outlined in Regulatory Guide 1.7. The time dependent hydrogen volume fraction was determined from the calculated hydrogen production and the known quantities of other noncondensables and water vapor comprising the postaccident containment atmosphere.

14.8.3.2.1 Radiolysis of Water

The postaccident fission product distribution outlined in Regulatory Guide 1.7 was used to calculate the hydrogen production due to radiolysis. All noble gases were considered to be released to the containment atmosphere, 50 percent of the halogens and 1 percent of the solids were considered to be released to the coolant, and the remaining fission products were considered to remain in the core. The decay energy release rate for solids was derived from the ANS-56.1 draft standard for two-year reactor operation as presented in the Standard Review Plan Section 6.2.5. The source inventories for all halogen and noble gas isotopes were incorporated in the radiolysis equation. This equation also accounted for the decay of the radioactive materials. (98-197)

Radiolysis of water was considered for the coolant adjacent to the core and in the sump. Gamma energy from in-core fission products, attenuated by a factor of 0.1, was used to calculate radiolysis of the coolant adjacent to the core. In addition, gamma and beta energy from the released halogens and solids was used to calculate radiolysis of the coolant in the core region and in the sump. In all cases, a constant $G(H_2)$ value of 0.5 molecules/100ev absorbed was used in the calculation. The integrated energy absorbed by the core and sump water following the design basis accident used in the calculation of hydrogen generation from radiolysis is shown in Figure 14.8.3-6. 99-92

14.8.3.2.2 Zirconium Water Reaction

Zirconium will react with steam according to the reaction

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2$$

The analysis assumes that there are 53,700 pounds of zircaloy clad in the core. Of this quantity, five percent was assumed to react after the postulated accident. This reaction produced 59.07 lbm-moles of free hydrogen. The hydrogen produced in this manner was assumed to be released to the containment atmosphere in the first few minutes after the accident occurred. (98-197)

14.8.3.2.3 Corrosion of Containment Metals

Free hydrogen may be liberated due to the corrosive effects on metals of containment spray solutions in conjunction with elevated containment temperatures. The metals whose corrosion contributes significantly to hydrogen production are aluminum and zinc. There is approximately 2,300 ft^2 of aluminum surface which may be exposed to the spray solutions. The corrosion rate for aluminum is approximately 200 mils per year.

There is approximately 125,100 ft^2 of galvanized steel surface in the containment. The corrosion rate for galvanized steel varies as a function of spray temperature, spray chemistry, and spray pH. The spray for Millstone is a borated solution with a pH of approximately 5.5. The temperature of the spray after equilibrium will vary throughout an assumed accident between 120°F and about 289°F. The following equation has been used to determine the effects of corrosion on the galvanized material (Reference 14.8-4). (98-197)

(98-197)

Corrosion Rate = $5.577 \times 10^9 e^{(-13142/T + 460)}$ mils/year. T is temperature (°F)

In addition to the corrosion of aluminum and galvanized steel, hydrogen may be produced by the reaction of the spray solution with zinc based paint. There is approximately 215,500 ft^2 of painted surface in the Millstone containment. The hydrogen production rate from the paint was based on ANS-56.1 draft standard for two-year reactor operation (Reference 14.8-1). The hydrogen generation rate for zinc based paint is as follows: (98-197)

Hydrogen Generation Rate = $0.3236 * e^{(-13134.6/T + 460)}$ lbm-mole/ft^2-sec. T is temperature (°F)

14.8.3.2.4 Time Dependent Hydrogen Concentration

The volume fraction of hydrogen as a function of time was calculated from the known quantities of noncondensables and the water vapor present in the containment atmosphere. Volume fractions were considered to be equal to the corresponding mole fractions. The total quantity of hydrogen produced from each source is plotted in Figure 14.8.3-1 for sixty days after the postulated accident. The total hydrogen production rate is shown in Figure 14.8.3-2. Finally, the hydrogen volume fraction is plotted in Figure 14.8.3-3.

The lower flammability limit of hydrogen is taken to be four (4) volume percent. In order to ensure a conservative safety margin, hydrogen recombiners will be started no later than the time at which the hydrogen concentration reaches three (3) percent. As is shown in Figure 14.8.3-3, this limit is reached at approximately 5.5 days after the accident.

As shown in Figure 14.8.3-2, the hydrogen production rate at the time when the hydrogen concentration reaches three (3) volume percent is 0.49 lbm-moles/hr. Since the production rate continuously decreases, the maximum required recombination rate is 0.49 lbm-moles/hr. This corresponds to a recombiner flow rate of 104 scfm. This is greater than the flow rate of one recombiner but within 6 hours hydrogen generation has decreased to the point where the rate of removal is greater than the rate of hydrogen production.

14.8.3.3 Hydrogen Concentration with One Recombiner in Operation

Since the analysis of hydrogen accumulation of Subsection 14.8.3.2 shows that the lower flammability level of four (4) volume percent will be exceeded, a thermal-type hydrogen recombiner system will be utilized to control the concentrations within safe limits.

The performance of the recombiners has been evaluated with respect to time of initiation. The following sections discuss the results of this study.

14.8.3.3.1 Proposed Operation

The preferred starting time of the recombiner is one day following a LOCA. It is felt that this starting time will provide adequate time for the operator to start the recombiners manually and also provide a large safety factor by limiting the peak hydrogen concentration to 2.63 volume percent, 1.37 volume percent below that allowed by Regulatory Guide 1.7. Figure 14.8.3-4 shows the effect of the recombiner, started one day after a LOCA, on the hydrogen concentration inside the containment.

14.8.3.3.2 Initiation Time

The starting of the recombiners could be delayed as long as 9.5 days and still maintain the hydrogen concentration below the four (4) volume percent limit specified by Regulatory Guide 1.7. The peak hydrogen concentration for various recombiner starting times are shown in Figure 14.8.3-5.

If a LOCA were to occur, one recombiner would be started one day after the occurrence, and, if a malfunction were discovered, the other recombiner would be activated. (99-67)

TABLE 14.6.5.1-3

MILLSTONE UNIT 2 SYSTEM ANALYSIS PARAMETERS
(LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS)

Primary Heat Output, MWt	2700*	
Primary Coolant Flow Rate, lbm/hr	1.36×10^8 (360,000 gpm)	
Primary Coolant System Volume, ft^3	11,000**	(98-200)
Operating Pressure, psia	2250	
Inlet Coolant Temperature, °F	549	
Reactor Vessel Volume, ft^3	4538	
Pressurizer Total Volume, ft^3	1500	
Pressurizer Liquid Total, ft^3	800	
SIT Total Volume, ft^3 (one of four)	2019	
SIT Liquid Volume, ft^3	1150.5	
SIT Pressure, psia	238.5	
SIT Fluid Temperature, °F	106.8	
Total Number of Tubes per Steam Generator	8523	
Steam Generator Tube Plugging	5.9%	(98-200)
Number of Tubes Plugged (Broken Loop)	500	
Number of Tubes Plugged (Double Intact Loop)	500	(98-200)
Steam Generator Secondary Side Heat Transfer Area (Broken Loop), ft^2	87,130	(98-200)
Steam Generator Secondary Side Heat Transfer Area, (Intact Loop), ft^2	87,130	(98-200)

*Primary Heat Output used in RELAP4-EM Model - 1.02 x 2700 = 2754 MWt.

**Includes pressurizer total volume and 5.9% SGTP. (98-200)

TABLE 14.6.5.1-3

MILLSTONE UNIT 2 SYSTEM ANALYSIS PARAMETERS
(LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS)

Steam Generator Secondary Flow Rate lbm/hr	6.04×10^6	(98-200)
Steam Generator Secondary Pressure (broken loop), psia	878.4	(98-200)
Steam Generator Secondary Pressure (intact loop), psia	878.4	(98-200)
Steam Generator Feedwater Temperature, °F	435	
Reactor Coolant Pump Rated Head, ft	271.8	
Reactor Coolant Pump Head, ft (DIL)	230.38***	
Reactor Coolant Pump Head, ft (SIL,BL)	233.00***	
Reactor Coolant Pump Rated Torque, ft-lbf	31,560	
Reactor Coolant Pump Rated Speed, rpm	892	
Initial Reactor Coolant Pump Speed, rpm	866.8***	(98-200)
Reactor Coolant Pump Moment of Inertia, lbm-ft^2	100,000	
Maximum Containment Net Free Volume, ft^3	1.938×10^6	
Containment Temperature, °F	101.6	(98-200)
SIS Fluid Temperature, °F	72.8	
HPSI Delay Time, sec	25.0	
LPSI Delay Time, sec	45.0	(98-200)

***Values used in RELAP4 for initialization.

TABLE 14.6.5.1-3

MILLSTONE UNIT 2 SYSTEM ANALYSIS PARAMETERS
(LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS)

SIS Delivery Curves for Loss-of-Diesel Single Failure****

RCS Pressure (psia)	DIL HPSI (gpm)	SIL HPSI (gpm)	BL HPSI (gpm)	DIL LPSI (gpm)	SIL LPSI (gpm)	BL LPSI (gpm)
1144.34	0.00	0.00	0.00	0.00	0.00	0.00
1100.00	61.38	30.69	30.69	0.00	0.00	0.00
1050.00	87.75	43.88	43.88	0.00	0.00	0.00
1000.00	108.65	54.33	54.33	0.00	0.00	0.00
900.00	137.89	68.95	68.95	0.00	0.00	0.00
700.00	178.35	89.18	89.18	0.00	0.00	0.00
500.00	213.13	106.57	106.57	0.00	0.00	0.00
300.00	243.43	121.22	121.22	0.00	0.00	0.00
200.00	256.67	128.34	128.34	0.00	0.00	0.00
150.00	263.57	131.31	131.59	0.00	498.35	605.38
100.00	270.04	133.80	134.54	0.00	881.92	1039.03
50.00	276.41	136.26	137.44	0.00	1156.35	1349.84
14.70	280.83	138.01	139.47	0.00	1310.58	1524.66
0.00	280.83	138.01	139.47	0.00	1310.58	1524.66

****SIS delivery to specific loops was chosen to ensure conservative results and thus does not reflect the actual plant cold leg/SIS train arrangement. For example, the larger of the two LPSI flows under loss-of-diesel conditions was directed to the broken loop. The model is insensitive to *intact* loop/SIS train assignments. Analysis delivery curves shown above differ from the latest calculated values. An evaluation has shown that the analysis values are bounding. 99-59

TABLE 14.6.5.1-3

MILLSTONE UNIT 2 SYSTEM ANALYSIS PARAMETERS
(LARGE BREAK LOSS OF COOLANT ACCIDENT ANALYSIS)

SIS Delivery Curves for Loss-of-LPSI Single Failure*****

RCS Pressure (psia)	DIL HPSI (gpm)	SIL HPSI (gpm)	BL HPSI (gpm)	DIL LPSI (gpm)	SIL LPSI (gpm)	BL LPSI (gpm)
1144.34	0.00	0.00	0.00	0.00	0.00	0.00
1100.00	122.57	61.13	61.32	0.00	0.00	0.00
1050.00	175.22	87.38	87.65	0.00	0.00	0.00
1000.00	216.94	108.18	108.52	0.00	0.00	0.00
900.00	275.49	137.38	137.81	0.00	0.00	0.00
700.00	356.28	177.66	178.22	0.00	0.00	0.00
500.00	425.78	212.31	212.98	0.00	0.00	0.00
300.00	484.25	241.46	242.23	0.00	0.00	0.00
200.00	512.67	255.64	256.45	0.00	0.00	0.00
150.00	525.61	262.00	263.11	653.94	274.77	398.76
100.00	537.51	267.76	269.33	1155.32	512.61	664.06
50.00	549.31	273.50	275.44	1475.25	663.50	834.75
14.70	557.54	277.50	279.70	1669.90	755.14	938.85
0.00	557.54	277.50	279.70	1669.90	755.14	938.85

(98-200)

*****SIS delivery to specific loops was chosen to ensure conservative results and thus does not reflect the actual plant cold leg/SIS train arrangement. For example, the largest of the LPSI flows under loss-of-LPSI conditions was directed to the broken loop. The model is insensitive to *intact* loop/SIS train assignments.

TABLE 14.6.5.2-3

MILLSTONE UNIT 2 SMALL BREAK LOSS OF
COOLANT ACCIDENT SYSTEM ANALYSIS PARAMETERS

Primary Heat Output, MWt	2,700*	
Primary Coolant Flow Rate, lbm/hr	1.36×10^8 (360,000 gpm)	
Primary Coolant System Volume, ft^3	10,997**	(98-200)
Operating Pressure, psia	2,250	
Inlet Coolant Temperature, °F	549	
Reactor Vessel Volume, ft^3	4,535	(98-200)
Pressurizer Total Volume, ft^3	1,500	
Pressurizer Liquid Total, ft^3	918	(98-200)
SIT Total Volume, ft^3 (one of four)	2,019	
SIT Liquid Volume, ft^3	1,150.5	
SIT Pressure, psia	215.0	
SIT Fluid Temperature, °F	106.8	
Total Number of Tubes per Steam Generator	8,523	
Number of Tubes Plugged per Steam Generator	500 (5.87%)	
Steam Generator Secondary Flow Rate, lbm/hr	6.01×10^6	(98-200)
Steam Generator Secondary Pressure, psia	830	
Steam Generator Feedwater Temperature, °F	435	
Reactor Coolant Pump Rated Head, ft	271.8	
Reactor Coolant Pump Rated Torque ft-lbf	31,560	
Reactor Coolant Pump Rated Speed, rpm	892	

*Primary heat output used in SPC-RELAP model - 1.02 x 2700 - 2754 MWt.

**Includes pressurizer total volume and 5.87% average SGTP.

TABLE 14.6.5.2-3

MILLSTONE UNIT 2 SMALL BREAK LOSS OF
COOLANT ACCIDENT SYSTEM ANALYSIS PARAMETERS

Initial Reactor Coolant Pump Speed, rpm	852.2***	(98-200)
Reactor Coolant Pump Moment of Inertia, lbm-ft^2	100,000	
SIS Fluid Temperature, °F	100.0	
Reactor Scram Low Pressure Setpoint, psia +	1,700	
SIAS Activation Setpoint Pressure, psia	1,500	
Secondary Safety Valve Setpoint, psia (Nominal) (3% Uncertainty in Lift Pressure was Assumed in the Analysis)	1,050	(98-200)
HPSI Delay Time, sec.	55.0	
SG Feedwater Piping Volume for AFW Sweepout, ft^3	141 + +	

HPSI Flow Rate vs. RCS Backpressure Data

RCS Pressure (psia)	HPSI Flow + + + (One Pump) (gpm)	
1204	0.00	99-59
1200	34.02	
1150	126.54	
1100	178.35	
1050	214.61	
1000	242.76	
900	292.88	(98-200)
700	379.30	
500	450.41	
300	515.77	
200	545.81	

***Value used in ANF-RELAP for initialization.

+The assumed value of 1700 psia conservatively bounds the actual value of 1865 psia, accounting for harsh containment conditions. (98-184)

+ +A sweepout volume of 152 ft^3 was determined to have insignificant effect on the analysis. (98-184)

+ + +Calculated values are slightly lower than analysis values. An evaluation has shown that the difference has insignificant effect on analysis results. 99-59

TABLE 14.6.5.2-3

MILLSTONE UNIT 2 SMALL BREAK LOSS OF
COOLANT ACCIDENT SYSTEM ANALYSIS PARAMETERS

Degraded AFW Flow Rate Versus Steam Generator Pressure
(1 Motor-Driven Pump)**** 99-59

SG Pressure (psia)	Flow to SG 1 (gpm)	Flow to SG 2 (gpm)
0	173	167
900	173	167
940	158	152
1030	115	111
1060	97	93
1060.1	0	0

(98-200)

Degraded AFW Flow Rate Versus Steam Generator Pressure
(1 Motor-Driven and Turbine Driven Pump)**** 99-59

SG Pressure (psia)	Flow to SG 1 (gpm)	Flow to SG 2 (gpm)
0	300	300
900	300	300
940	300	300
1030	236	227
1060	193	186
1060.1	0	0

****Calculated values differ from the analysis values. An evaluation has shown that the calculated values do not adversely impact analysis results. 99-59

TABLE 14.6.5.2-3

MILLSTONE UNIT 2 SMALL BREAK LOSS OF
COOLANT ACCIDENT SYSTEM ANALYSIS PARAMETERS

Degraded AFW Flow Rate Versus Steam Generator Pressure (2 Motor-Driven Pumps)**** 99-59

SG Pressure (psia)	Flow to SG 1 (gpm)	Flow to SG 2 (gpm)
0	294	283
900	294	283
940	269	259
1030	195	188
1060	164	158
1060.1	0	0

(98-200)

****Calculated values differ from the analysis values. An evaluation has shown that the calculated values do not adversely impact analysis results. 99-59

TABLE 14.8.2-4

SEQUENCE OF EVENTS
MP2-MSLB: FAILURE OF VITAL BUS VA-10 OR VA-20 TO ENERGIZE FROM 0 PERCENT POWER

TIME (seconds)	EVENT	SETPOINT/VALUE
0.00	MSLB occurs from 0% power, break size is 1.75 ft^2.	
0.00	Maximum AFW flow to the affected SG.	660 gpm, 100°F
4.85	Containment High Pressure signal (CHPS) is generated. This will cause a Reactor Trip and MSIS after a 0.9 second signal delay.	5.83 psig with uncert.
5.75	Reactor Trip and MSIS occurs.	
11.58	Containment High-High Pressure signal (CHHPS) occurs.	11.08 psig with uncert.
19.85	Containment cooling fans energize. Time based on CHPS + 15 second delay.	
60.58	Containment spray flow commences. Time based on CHHPS + 49.0 seconds for pump start, valve stroke time, and header fill time.	1300 gpm
60.58	Peak Containment Temperature reached.*	377.6°F
274.18	Peak Containment Pressure reached.*	53.7 psig
600.00	Simulation ended.	

99-73

*This case has been reanalyzed considering the affect of increased paint thickness on the heat structures inside containment. This case results in a peak containment temperature of 377.9°F at 60.58 seconds and a peak containment pressure of 53.8 psig at 273.1 seconds. 99-73

TABLE 14.8.4-2

SUMMARY OF DOSES FOR LOSS OF COOLANT ACCIDENT

DOSE (rems)

	Thyroid	Whole Body
EAB	3.46 E+01	2.36 E+00
LPZ	1.31 E+01	9.08 E-01

(98-145)

98-145 ADMIN

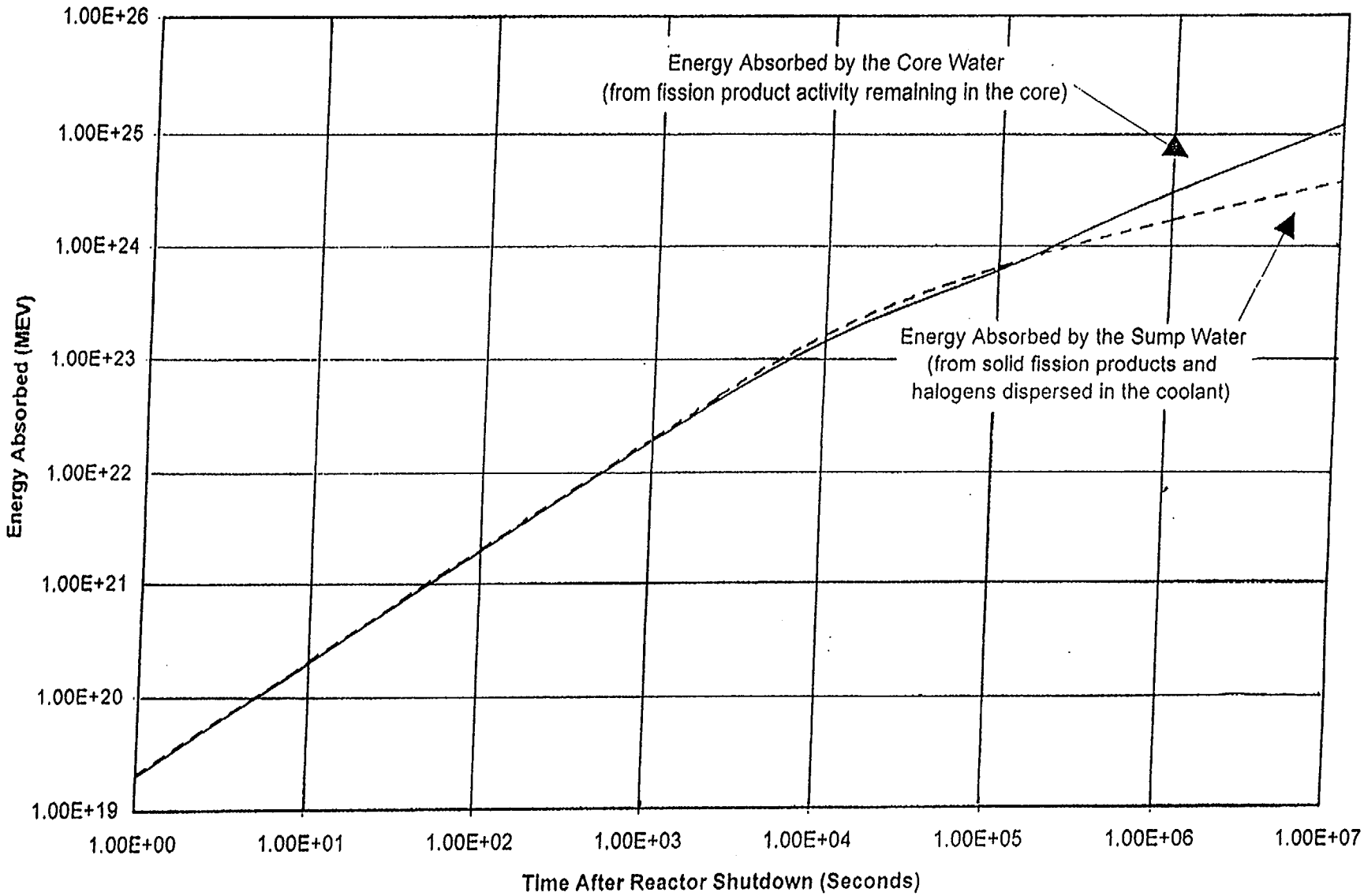

FIGURE 14.8.3-6
FISSION PRODUCT ENERGY ABSORBED BY THE COOLANT FOLLOWING A LOSS OF COOLANT ACCIDENT